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ABSTRACT

This report documents the U.S. Nuclear Regulatory Commission (NRC) staff review and safety and safeguards evaluation of the United States Enrichment Corporation, Inc. (USEC, the applicant) application for a license to possess and use special nuclear material (SNM) and source material in the American Centrifuge Lead Cascade Facility (Lead Cascade). The Lead Cascade is a test and demonstration facility to be located at the Portsmouth Gaseous Diffusion Plant (PORTS) in Piketon, Ohio. The facility will possess up to 250 kilograms of uranium hexafluoride (UF_6) and will consist of up to 240 operating, full-scale centrifuge machines arranged in a cascade configuration.

The objective of this review is to evaluate the potential adverse impacts of operation of the facility on worker and public health and safety under both normal operating and accident conditions. The review also considers physical protection of SNM and classified matter, material control and accounting of SNM, and the management organization, administrative programs, and financial qualifications provided to assure safe design and operation of the facility.

The NRC staff concludes in this safety evaluation report (SER) that the applicant's descriptions, specifications, and analyses provide an adequate basis for safety and safeguards of facility operations and that construction and operation of the facility does not pose an undue risk to worker and public health and safety.

EXECUTIVE SUMMARY

On February 11, 2003, United States Enrichment Corporation Inc. (the applicant), submitted to the U.S. Nuclear Regulatory Commission (NRC), an application requesting a license under 10 CFR Part 70 to possess and use special nuclear, source and by-product material in the American Centrifuge Lead Cascade Facility (Lead Cascade).

The Lead Cascade is a test and demonstration facility that will be located at the Portsmouth Gaseous Diffusion Plant in Piketon, Ohio. The facility will possess up to 250 kilograms of uranium hexafluoride (UF_6) and will consist of up to 240 operating, full-scale centrifuge machines arranged in a cascade configuration. The cascade will be operated in recycle mode, i.e., no enriched product will be withdrawn. The only withdrawal of enriched UF_6 from the cascade will be in the form of samples.

The NRC staff conducted its safety review in accordance with NUREG-1520 "Standard Review Plan for the Review of a license application for a Fuel Cycle Facility." The staff's safeguards review involved reviews of the applicant's Fundamental Nuclear Material Control (FNMC) Plan and Physical Protection Plan for Special Nuclear Material (SNM) and Classified Matter.

A summary of NRC's review and findings in each of the review areas are provided below:

General Information

The applicant provided an adequate description of the facility and processes so that the staff has an overall understanding of the relationships of the facility features as well as the function of each feature. Financial qualifications were properly explained and outlined in the application package. The description of the site included important information about regional meteorology, hydrology, geology, nearby population, and potential effects of natural phenomena on the facility.

Organization and Administration

The applicant adequately described the responsibilities and associated resources for the design, construction, and operation of the facility and its plans for managing the project. The plans and commitments described in the application provide reasonable assurance that an acceptable organization, administrative policies, and sufficient competent resources have been established or committed to satisfy the applicant's commitments for the design, construction, and operation of the facility.

Integrated Safety Analysis (ISA) and ISA Summary

The applicant provided sufficient information about the site, facility processes, hazards, and types of accident sequences. The information provided addressed each credible event, the potential radiological and chemical consequences of the event, and the likelihood of the event. No mitigated event consequence exceeds the performance requirements at 10 CFR 70.61. The applicant also provided adequate information about the items relied on for safety (IROFS).

Radiation Protection

The applicant provided sufficient information to evaluate the Radiation Protection Program. The application adequately describes the qualification requirements, written radiation protection procedures and Radiation Work Permits (RWPs), and necessary training for all personnel who have access to radiologically restricted areas. The radiation survey and monitoring program is adequate to protect the workers and any other persons that may be potentially exposed to radiation.

Nuclear Criticality Safety

The applicant provided adequate information to evaluate the Nuclear Criticality Safety (NCS) program. The applicant has committed to having an adequate group of qualified staff to develop, implement, and maintain the NCS program in accordance with the facility organization and administration and management measures. The program will meet the regulatory requirements for the margin of subcriticality for safety.

Chemical Process Safety

The applicant has adequately described and assessed accident consequences that could result from the handling, storage, or processing of licensed materials and that could have potentially significant chemical consequences and effects. The applicant has constructed hazard analyses that identified and evaluated those chemical process hazards and potential accidents and established safety controls, which meet the regulatory requirements.

Fire Safety

The applicant commits to reasonable administrative controls and engineered controls to minimize the risk of fires and explosions. The IROFS and defense-in-depth protection discussed in the applicant's ISA Summary, along with safety basis assumptions as described, and the planned programmatic commitments in the license application provide reasonable protection against a fire hazard and meet safety requirements.

Emergency Management

The applicant provided an adequate emergency plan for the facility that meets the regulatory requirements. The applicant commits to maintaining and executing an emergency plan for responding to the radiological and chemical hazards resulting from a release of radioactive material or hazardous chemicals incident to the processing of licensed material. The requirements of the emergency plan are implemented through approved written procedures.

Environmental Protection

The applicant has committed to adequate environmental protection measures, including (1) environmental and effluent monitoring and (2) effluent controls to maintain public doses as low as reasonable achievable (ALARA) as part of the radiation protection program. USEC's proposed controls are adequate to protect the environment and the health and safety of the public

and comply with the regulatory requirements.

Decommissioning

The applicant provided sufficient information to demonstrate that the decommissioning plan for the Lead Cascade facility is adequate and will not pose a threat to public health and safety or the environment. The applicant provided adequate information to conclude that there will be sufficient funds and financial assurance available at the time of decommissioning. The information is acceptable to conclude that the Decommissioning Plan meets the regulatory requirements.

Management Measures

The applicant provided adequate information about the overall configuration management (CM) program and policy. The application properly described the maintenance program, training, and the process for the development, approval, and implementation of procedures. The applicant explained the audits and assessments program as well as incident investigations and records management system. The applicant has committed to establishing and documenting surveillances, tests, and inspections to provide reasonable assurance of satisfactory performance of IROFS. The information is acceptable to conclude that the CM program and policy meet the regulatory requirements.

Materials Control and Accountability

The applicant provided acceptable information that describes the Fundamental Nuclear Material Control (FNMC) Plan for the Lead Cascade facility. The plan will meet or exceed the applicable regulatory requirements to control and account for special nuclear material that THE APPLICANT will use, possess, or have access to.

Physical Protection

SNM

The applicant provided sufficient information about the specific policies, methods, and procedures to be implemented to protect the special nuclear material of low strategic significance in the Lead Cascade facility. The information is acceptable to conclude that the Physical Protection Plan meets the regulatory requirements.

Classified Matter

The security plan for the protection of classified matter outlines the facility's proposed security procedures and controls to ensure that classified matter is used, processed, stored, reproduced, transmitted, transported, and destroyed in accordance with the regulatory requirements.

ACRONYMS AND ABBREVIATIONS

ACL	administrative control level
AIHA	American Industrial Hygiene Association
ALARA	as low as reasonable achievable
ALOHA	areal locations for hazardous atmospheres
amsl	above mean sea level
ANS	American Nuclear Society
ANSI	American National Standard Institute
ARA	airborne radioactivity area
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
BDC	baseline design criteria
BEQ	baseline effluent quantity
BMP	best management practices
CA	contamination area
CAA	controlled access area
CEDE	committed effective dose equivalent
CFR	Code of Federal Regulations
cfs	cubic feet per second
Ci	Curie
CM	crisis manager configuration management
CMP	classified matter plan
CRR	cyclic resistance ratios
CSR	cyclic stress ratio
CTTF	centrifuge training and test facility
DAC	derived air concentration
DCD	Design Criteria Documents
DFP	Decommissioning Funding Plan
DOE	Department of Energy
DOT	Department of Transportation
DR	damage ratio
EIS	Environmental Impact Statement
EM	emergency management
EOC	Emergency Operations Center
EPA	Environmental Protection Agency
EPIP	Emergency Plan Implementing Procedures
ERPG	Emergency Response Procedure Guideline
EV	evacuation vacuum
ft	feet
FNAD	fixed nuclear accident dosimeters
FNMC	fundamental nuclear material control
fpm	feet per minute
FSRC	Facility Safety Review Committee
g	acceleration of gravity
GCEP	Gas Centrifuge Enrichment Program

GDP	Gaseous Diffusion Plant
h	hours
HCA	high contamination area
HEPA	high-efficiency particulate air
HP	Health Physics
HRA	high radiation area
HS&E	health, safety, and environment
Hz	Hertz
IC	incident commander
IHS	industrial hygiene and safety
in.	inches
ICP	inductively couple plasma
IPT	intraplant transporter
IROFS	item(s) relied on for safety
ISA	integrated safety analysis
ISO	International Organization for Standardization
kg	kilograms
km	kilometers
kPa	kilopascals
LA	license application
LC	Lead Cascade
LEC	liquid effluent collection
LPF	leak path factor
m	meters
MAR	material-at-risk
MC&A	material control and accountability
MCW	machine cooling water
MDC	minimum detectable concentration
MOU	Memorandum of Understanding
MOX	mixed oxide
mph	miles per hour
MS	mass spectrometry
MT&E	measuring and test equipment
NCS	nuclear criticality safety
NCSA	nuclear criticality safety approval
NCSE	nuclear criticality safety evaluation
NESHAP	National Emission Standards for Hazardous Air Pollutants
NFPA	National Fire Protection Association
NIST	National Institute of Standards and Technology
NPDES	National Pollution Discharge Elimination System
NRC	Nuclear Regulatory Commission
NS&Q	nuclear safety and quality
NVLAP	National Voluntary Laboratory Accreditation Program
OJT	on-the-job training
OSHA	Occupational Safety and Health Administration

PBT	performance based training
PEL	permissible exposure limits
PHA	process hazard analysis
PM	preventive maintenance
PMF	probably maximum flood
PORTS	Portsmouth Gaseous Diffusion Plant
PP	physical protection
PPE	personal protection equipment
PSB	process support building
psf	pounds per square foot
PSM	process safety management
PSS	plant shift superintendent
PV	purge vacuum
QA	quality assurance
QAPD	quality assurance program description
QC	quality control
QL	quality level
RA	radiation area
R/A	recycle/assembly
RCW	recirculating cooling water
REIRS	radiation exposure information reporting system
RM	river mile
RMDC	Records Management and Document Control
RMP	Risk Management Program
RP	radiation protection
RPM	radiation protection manager
RWP	radiation work permit
SEC	Securities and Exchange Commission
SER	safety evaluation report
SNM	special nuclear material
SNM-LSS	special nuclear material of low strategic significance
SPCC	spill prevention control and countermeasure
SRD	system requirements documents
SSCs	structures, systems and components
ST	source term
STP	sewage treatment plan
TEDE	total effective dose equivalent
TRM	training requirement matrices
UF_6	uranium hexafluoride
USEC	United States Enrichment Corporation
WCA	worker in the controlled area
WI/CL	what-if/checklist
WRA	worker in the restricted area

1.0 GENERAL INFORMATION

1.1 FACILITY AND PROCESS DESCRIPTION

The purpose of the Nuclear Regulatory Commission's (NRC's) review of the facility and process description is to ascertain whether an application includes an overview of the facility layout and a summary description of its manufacturing processes. A more detailed description of the facility and its manufacturing processes is contained in the integrated safety analysis (ISA) Summary.

1.1.1 REGULATORY REQUIREMENTS

The regulations in 10 CFR 70.22 require each application to include information on the proposed activity, and the equipment and facilities which will be used by the applicant to protect health and minimize danger to life and property. In addition, the regulations in 10 CFR 70.65 require each application to include a general description of the facility with emphasis on those areas that could affect safety, including identification of the controlled area boundaries.

1.1.2 REGULATORY ACCEPTANCE CRITERIA

The acceptance criteria applicable to the NRC's review of the facility and process description section of the application are contained in Section 1.1.4.3 of the Standard Review Plan for Fuel Cycle Facilities (NUREG-1520).

1.1.3 STAFF REVIEW AND ANALYSIS

In Section 1.1 of the application, United States Enrichment Corporation (The applicant) describes its Lead Cascade facilities and process description. The Lead Cascade facilities include the X-3001 (PB1) Process Building, which houses up to 240 operating centrifuge machines, associated process piping, instrumentation and controls, computer systems, and auxiliary support equipment. The facilities also include the X-3012 Process Support Building (PSB) to provide oversight and control of the equipment in the cascade. The X-7726 Centrifuge Training and Test Facility (CTTF) provide areas to receive and test centrifuge components, and to assemble and repair the centrifuges. An intraplant transporter (IPT) moves centrifuge machines between the CTTF and PB1 through the covered X-7727H Transfer Corridor. The X-3012 also provides offices, lockers, change rooms, and break rooms. A portion of the X-7725 Recycle/Assembly (R/A) Building provides similar administrative facilities, as well as training rooms, and the storage and maintenance areas for the IPT. Support facilities for the Lead Cascade include facilities for emergency response, training, maintenance, laboratory support, utilities, environmental and waste management support, and administrative support. These are provided through existing facilities at the Portsmouth Gaseous Diffusion Plant (PORTS). The application provides additional descriptions and process details for each of these areas.

1.1.4 EVALUATION FINDINGS

The staff has reviewed the Lead Cascade's facility and process descriptions according to Section

1.1 of the Standard Review Plan. The applicant has adequately described (1) the facility and processes so that the staff has an overall understanding of the relationships of the facility features and (2) the function of each feature. The staff concludes that the applicant has met the requirements and acceptance criteria applicable to this section.

1.2 INSTITUTIONAL INFORMATION

The purpose of the NRC's review of institutional information is to establish whether the license application includes adequate information identifying the applicant, the applicant's characteristics, and the proposed activity.

1.2.1 REGULATORY REQUIREMENTS

The regulations in 10 CFR 70.22 require each application to include information on corporate identity, facility location, other site activities, financial qualifications, and the name, amount, and specifications of the special nuclear material (SNM) to be used. In addition, the applicant has to have requested and received a facility security clearance in accordance with 10 CFR Part 95.

1.2.2 REGULATORY ACCEPTANCE CRITERIA

The acceptance criteria applicable to the NRC's review of the institutional information section of the application are contained in Section 1.2.4.3 of the Standard Review Plan for Fuel Cycle Facilities (NUREG-1520).

1.2.3 STAFF REVIEW AND ANALYSIS

A) Corporate Identity

In Section 1.2.1 of the application, the applicant discusses its corporate identity by stating that the applicant, including its wholly owned subsidiaries, was organized under Delaware law in connection with the privatization of the Corporation. Section 1.2.1 identifies the principal officers of the applicant and provides information on the locations of the applicant's principal office and the Lead Cascade facility. Section 1.2.1.2 generally describes other ongoing operations at PORTS which includes:

1. Maintaining PORTS in Cold Standby status under a contract with the Department of Energy (DOE);

2. Performing uranium deposit removal activities in the gaseous diffusion cascade facilities; and

3. Removing technetium (^{99}Tc) from potentially contaminated uranium feed in accordance with the June 17, 2002, agreement between the applicant and the DOE.

Section 1.2.1.2 also states that the applicant possesses a license for radioactive material operations from the State of Ohio for laboratory and associated support activities. This license encompasses laboratory analyses, in-field analyses for radioactive material deposits, and health

physics survey and characterization activities. In addition to the applicant's operations, the DOE is engaged in environmental restoration activities in a number of locations on the reservation and utilizes contractors and sub-contractors to perform this work. DOE self-regulates these activities. Additionally, the Ohio National Guard maintains an area on the PORTS reservation for the maintenance, reconditioning, and storage of equipment. No ordnance is permitted. The activities are accomplished in and around the X-751 facility, located on the south end of the site.

B) Financial Qualifications

The staff evaluated proprietary financial qualification information provided by the applicant by reviewing Section 1.2 of the application and proprietary supplementary information provided by the applicant (Ref. 1.2.5.3).

i) Project Costs

The applicant provided an estimate of the project costs to build and operate the proposed Lead Cascade facility. The applicant plans to self-finance the facility or not seek external funding. No other financial partners or debt financing are expected for the project. The applicant has indicated that it would fund the Lead Cascade project from existing and future cash flows from ongoing operations. In the event of cost overruns or funding shortfalls, the applicant would also provide funding from existing and future cash flows from on-going operations.

In Appendix D to the Lead Cascade Environmental Report, the applicant provided an estimate of the project costs. Projected costs included costs for process machinery and equipment; building and infrastructure; centrifuge machines; engineering, procurement, and construction; operation; and regulatory compliance.

ii) Financial Qualifications

The applicant is a publicly-held company and, therefore, is required to submit Report 10-K to the Securities and Exchange Commission (SEC). Reports 10-K are available for the applicant's fiscal year 2002 ending on June 30, 2002, dated September 20, 2002, and for a six-month fiscal year transition period ending December 31, 2002, dated March 4, 2003. The transition Report 10-K was prepared to reflect a corporate decision to change fiscal years from one ending on June 30 to one ending on December 31. The Reports 10-K contain balance sheets, cash flows, income statements, and equity statements. An independent accountant's report is also provided. The Independent Accountant concluded that it was not aware of any material modifications that should be made to the accompanying financial statements in order for them to be in conformity with accounting principles generally accepted in the United States. Information from the Report 10-K for fiscal year 2002, is also provided in the applicant's 2002 Annual Report.

The staff reviewed the SEC Reports 10-K and, considering future corporate obligations over the planned life of the project (including other gas centrifuge development costs) and the possible use of other available assets and financial instruments, determined that cash flow from current operations is sufficient to provide funding of the proposed project costs and any significant contingencies that could occur.

iii) Liability Insurance

As a laboratory-scale test and demonstration facility designed or used for experimental or analytical purposes only, under 10 CFR 140, public liability insurance is not required.

The staff reviewed the project costs, financial qualifications, and liability insurance plans for the Lead Cascade project according to Chapter 1.2 of NUREG-1520 and Chapter 2.0 of NUREG-1718. The staff used NUREG-1718 as it contains explicit guidance and acceptance criteria on financial qualifications not contained in NUREG-1520 that are applicable to the Lead Cascade facility. The staff evaluated estimates of project construction costs, sources of funds, contingencies, and financial qualifications, and found that these areas provide reasonable assurance that the applicant is qualified to properly construct and operate the proposed Lead Cascade facility with adequate funding.

C) Type, Quantity, and Form of Licensed Material

Table 1.2-1 of the application lists the type, quantity, and form of the Lead Cascade's possession limits for special nuclear, source, and by-product material. The 250 kilograms (kg) of UF_6 and 1.0 Curie (Ci) of thorium (to be used as instrument calibration and check sources and for nondestructive analyses) constitute the Lead Cascade's source material possession limits, while the 700 grams (g) of U-235 and 3.5 Ci of plutonium (to be used as instrument calibration and check sources and for nondestructive analyses) constitute the Lead Cascade's special nuclear material possession limits.

D) Authorized Uses

Table 1.2-2 of the application lists specific authorized uses of the source material and SNM. The Lead Cascade is not considered to be a production facility. Therefore, possession and use of byproducts and source material would fall under the purview of the State of Ohio, an Agreement State, while the NRC would regulate the SNM. Most of the source material that the applicant will possess and use will be an integral part of the process containing SNM. Since no distinct boundary exists between such source material and the SNM, the NRC will exercise regulatory jurisdiction over such source material. The State would regulate the remaining source material which will consist of calibration sources. As calibration sources containing byproducts and source material are needed for radiation safety, the applicant will need to obtain authorization from the State of Ohio to possess and use them. As such, the staff recommends the following license condition:

> Licensed activities at the American Centrifuge Lead Cascade Facility may not commence until authorization from the State of Ohio is obtained to use byproduct and source material needed to support safe nuclear operations at the facility.

Section 1.2.4 states that the Lead Cascade is operated on recycle where the enriched product stream is recombined with the depleted stream prior to being re-fed to the cascade. No product

withdrawals are made from the Lead Cascade except for samples taken for laboratory analysis. Withdrawal of small quantities of depleted material may also be performed on an infrequent basis for operational considerations, with subsequent addition of feed to the cascade.

E) Special Exemptions or Special Authorizations

The applicant has requested an exemption from the following 10 CFR Part 20 posting and labeling requirements:

- Five, eight, and twelve-inch uranium cylinders are routinely transported between facility locations and/or storage areas at the Lead Cascade and are readily identifiable due to their size and unique construction. These are not routinely labeled as radioactive material. The transportation of uranium hexafluoride (UF_6) cylinders and UF_6 sample containers are constantly attended by qualified Radiological Workers during movement.

The applicant has requested an exemption from the following 10 CFR 70.50 reporting requirement:

- Because of the comprehensive nature of event follow-up reports, the event analysis and root cause determinations are often not completed within 30 days. Thus, the initial 30-day report required by the regulations may be incomplete and a supplemental report must be prepared when the information is available. In recognition of this, follow-up written reports for the Lead Cascade are submitted within 60 days of an event, consistent with the exemption granted to the gaseous diffusion plants (GDPs) for reporting of events pursuant to 10 CFR 76.120(d)(2) (67 *Federal Register* 68699, November 12, 2002).

The applicant has requested the following Special Authorization:

- Surface Contamination Release Levels for Unrestricted Use – Items may be released for unrestricted use if the surface contamination is less than the levels listed in Table 4.6-1.

These requests are discussed in the pertinent sections of this Safety Evaluation Report.

F) Security of Classified Information

The NRC has reviewed and found acceptable the applicant's plan for the protection of classified matter (Chapter 2 of the Lead Cascade Security Program). NRC's review is described in Appendix E to this safety evaluation report.

1.2.4 EVALUATION FINDINGS

The applicant provided adequate information about: (1) corporate identity, including its full name and address, (2) financial qualification and resources, (3) quantity and characteristics of

licensed material to be used at the Lead Cascade facility, (4) authorized uses of the licensed material, and (5) security plan to protect classified information.

The NRC staff concludes, with reasonable assurance, that the institutional information provided by the applicant is acceptable and meets the regulatory requirements outlined in Section 1.2.1.

1.2.5 REFERENCES

1.2.5.1 Nuclear Regulatory Commission (U.S.) (NRC). NUREG-1520, "Standard Review Plan for the Review of a License application for a Fuel Cycle Facility," NRC: Washington, D.C. 2002.

1.2.5.2 Nuclear Regulatory Commission (U.S.) (NRC). NUREG-1718, "Standard Review Plan for the Review of an Application for a Mixed Oxide (MOX) Fuel Fabrication Facility." NRC: Washington, D.C. 2000.

1.2.5.3 Letter to U.S. Nuclear Regulatory Commission, RE Submittal of Additional Financial Qualification and Cost Estimate Information, August 29, 2003.

1.2.5.4 U.S. Securities and Exchange Commission (U.S.) (SEC). Report 10-K, "Annual Report Pursuant to Section 13 or 15 (d) of the Securities Exchange Act of 1934 USEC Inc." SEC: Washington, D.C. 2002.

1.2.5.5 U.S. Securities and Exchange Commission (U.S.) (SEC). Report 10-K, "Transition Report Pursuant to Section 13 or 15 (d) of the Securities Exchange Act of 1934 USEC Inc." SEC: Washington, D.C. 2003.

1.2.5.6 U.S. Enrichment Corporation (USEC). "Environmental Report American Centrifuge Lead Cascade Facility," Appendix D. USEC: Bethesda, MD 2003.

1.3 SITE DESCRIPTION

The purpose of the NRC's site description review is to determine whether the information provided by an applicant adequately describes the geographic, demographic, meteorological, hydrologic, geologic, and seismologic characteristics of the site and the surrounding area. The site description is a summary of the information that the applicant used in preparing the environmental report, emergency plan, and ISA Summary.

1.3.1 REGULATORY REQUIREMENTS

The regulations in 10 CFR 70.65(b)(1) require each application to include a general description of the site with those factors that could affect safety (i.e., meteorology, seismology).

1.3.2 ACCEPTANCE CRITERIA

The acceptance criteria applicable to the NRC's review of the site description section of the

application are contained in Section 1.3.4.3 of the Standard Review Plan for Fuel Cycle Facilities (NUREG-1520).

1.3.3. STAFF REVIEW AND ANALYSIS

A) Site Geography

Section 1.3 of the application describes the Lead Cascade's location and description, nearby roadways and bodies of water, and significant geographical features. Section 1.3 states that the Lead Cascade is located on DOE-owned land in rural Pike County, a sparsely populated area in south central Ohio. Specifically, the Lead Cascade is located on the PORTS reservation in the former Gas Centrifuge Enrichment Program (GCEP) facilities. The PORTS reservation is in Pike County on the east side of the Scioto River approximately equidistant between Portsmouth and Chillicothe, Ohio. The entire PORTS reservation is marked and bounded by signs and fences (barbed wire in the wooded areas). Where roads cross the boundary, gates are in place to serve as barriers if needed. PORTS reservation boundaries are identified in Figure 1.1-1 of the application. The reservation boundary is the controlled area boundary specified in 10 CFR 70.61(f). Most buildings and activities at the site (including the Lead Cascade facilities) are located within the next level of control, a Property Protection Area or Controlled Access Area (CAA), both surrounded by a security fence. Access to this fenced area is gained only with approved identification. In addition, the Lead Cascade is located within its own CAA. A topographic map of the PORTS reservation is provided in Figure 1.3-1 of the application.

With the exception of the Scioto River floodplain, which is farmed extensively, the area around the site consists of marginal farmland and forested hills. The only other body of water located near the site is Lake White, located approximately six miles north of the site.

B) Demographics

The nearest residential center and the closest town to the site is Piketon, located in Pike County, about four miles north of the site on U.S. Route 23, with a population of 1,907 in 2000. The largest town in Pike County is Waverly, about eight miles north of the site, with a population of 4,433 in 2000. The total population within the five-mile radius of the site is 5,836.

The two school systems in the area are the Pike County Schools and the Scioto County Schools. However, only Pike County has school facilities within five miles of the facility: two elementary schools, one that also has a preschool included; one high school; and a vocational school. The combined enrollment of these schools for the year 2002 was approximately 2,387. The total school population within five miles, including faculty and staff, is 2,681. The proximity of these schools to the site and their enrollments are shown in Figure 1.3-3 of the application.

Three facilities within five miles of the PORTS reservation provide day care or schooling for preschool-aged children and after-school care for school-aged children. One facility, licensed to accommodate 320 children, is located in Piketon; the other, licensed to accommodate 70 children, is located near the PORTS reservation boundary. The third facility is consolidated in the numbers provided above for the elementary school. The locations of these facilities are

shown in Figure 1.3-3 of the application.

Pike Community Hospital is the hospital closest to the site, located approximately 7.5 miles north of the facility on State Route 104 south of Waverly. The facility has 37 licensed beds and operates at full capacity. No other acute care facilities are located in Pike County. The location of Pike Community Hospital is shown in Figure 1.3-3. Adena Health Center operates an urgent care facility approximately 7.5 miles north of the site.

Two licensed nursing homes are located near Piketon and one in Wakefield; all are located within five miles of the site. The largest of these facilities is a 201-bed facility in Piketon.

No significant recreational areas are on the site; recreational activities for employees are held off site.

Off-site recreational areas include the Brush Creek State Forest, a 0.5 square mile portion of which is within five miles southwest of the PORTS reservation. Usage of this area is extremely light and is estimated to be 20 persons/year, primarily hunters and mushroom pickers. The location of Brush Creek State Forest is identified in Figure 1.3-3.

C) Meteorology

Section 1.3.3 of the application provides a meteorological description of the site and its surrounding area. Section 1.3.3.1 states that July is the hottest month, with an average monthly temperature of 74^{o}F, and January is the coldest month with an average temperature of 30^{o}F. The highest and lowest daily temperatures from 1951 to 1980 were 103 and –25^{o}F on July 14, 1954, and February 3, 1951, respectively. The average annual precipitation at Waverly, Ohio, for the period from 1951 to 1980 was 40.4 inches (in.). The average annual snowfall for the area is about 22 in., based on the 1951-1980 data. During that time period, the maximum monthly snowfall was 25.4 in., occurring in January 1978. The predominant winds at the site blow from the south or southwest and at times from the north. The average wind speed is about 5 miles per hour (mph). On the average, from 1953 to 1989, 14 tornadoes per year were reported in Ohio, but the total varies widely from year to year (e.g., 43 in 1973 and 0 in 1988). Pike County had two tornadoes during the 20-year period from 1953 to 1972.

i) Tornado Hazard and Tornado-Generated Missiles

Information about the tornadoes and design basis tornado at the Portsmouth Gaseous Diffusion Plant (PORTS) reservation area, where the proposed American Centrifuge Lead Cascade Facility is located, is provided in Section 1.3.3.3, "Local Meteorology," of the license application (Ref. 1.3.5.22) and Section 1.3.1, "Tornadoes and High Winds," of the redacted Integrated Safety Analysis Summary (Ref. 1.3.5.22).

According to information provided in the redacted Integrated Safety Analysis Summary, the State of Ohio had 235 tornadoes reported from 1953 to 1972. Among them, two tornadoes are known to have occurred in Pike County, where the proposed American Centrifuge Lead Cascade Facility is located (Ref. 1.3.5.23). The applicant presented a calculation in its Redacted

Integrated Safety Analysis Summary for determining the annual probability of tornado occurrence at the site. This calculation used a formula proposed by Fujita (Ref. 1.3.5.5) to determine the likelihood of a tornado at the site. The result indicated that the annual probability for a tornado at the site is approximately 8×10^{-4}, indicating that tornado hazard has to be considered in the design of the proposed facility and in the integrated safety analysis to assess the potential consequence of a tornado hazard. Based on the analysis result, the applicant concluded that the calculated tornado probability agrees with the severe winds and tornado hazard curves recommended for the Portsmouth, Ohio, area by Beavers, et al. (Ref. 1.3.5.3.)

In its license application (Ref. 1.3.5.22), the applicant indicates that the process building containing the lead cascade operations in the proposed American Centrifuge Lead Cascade facility is designed for a 100-mph [161-km/h] tornado and a rate of atmospheric pressure change of 7.0 $lb/ft^2/s$ [34 $kg/m^2/s$]. In the redacted Integrated Safety Analysis Summary, this tornado design basis is characterized as a 30,000-yr tornado. In response to staff questions raised during their site visit, the applicant (Ref. 1.3.5.17) indicated that this 30,000-yr tornado was determined using the tornado hazard curves recommended by Beavers, et al. (Ref. 1.3.5.3.) No design bases related to tornado-generated missiles are provided. The potential effect of tornado-generated missiles, however, is considered in the integrated safety analysis, and more discussion is provided later about the consideration of the potential effect of tornado-generated missiles on the process building.

Based on the high-wind and tornado hazard curves recommended by Beavers, et al. (Ref. 1.3.5.3) the staff determined that a 30,000-yr tornado should have a wind speed of approximately 110 mph [177 km/h]. This wind speed is consistent with the tornado speed recommended by Coats and Murray (Ref. 1.3.5.5). The design basis tornado wind speed used by the applicant for the process building is approximately 10 mph [16 km/h] lower than that recommended by Beavers, et al. (Ref. 1.3.5.3) and Coats and Murray (Ref. 1.3.5.5).

The applicant indicates in its redacted Integrated Safety Analysis Summary that "information on the ability of the facilities involved in the Lead Cascade Project to survive a tornado is not available." Consequently, the applicant has included an event scenario of tornado-induced damages to the process building and the centrifuges. This event scenario assumes that the process building will fail as a result of a tornado strike, and the centrifuge machines may be damaged by tornado-generated missiles. Based on this event scenario, the radiological and chemical consequences are estimated and reported in the redacted Integrated Safety Analysis Summary. The review of the appropriateness and acceptability of the methods and assumptions used for radiological and chemical release calculations are documented in Chapter 3 of this Safety Evaluation Report.

Because the difference is small between the tornado design basis used and that recommended, and the potential consequences caused by a tornado are estimated to be small, staff concludes that the tornado design basis used by the applicant is acceptable.

ii) High Winds and Hurricanes

Information about high winds at the proposed American Centrifuge Lead Cascade Facility is

provided in Section 1.3.3.3, "Local Meteorology," of the license application (Ref. 1.3.5.22).

According to the license application, the prevailing winds at the site are from south or southwest. The average wind speeds were 3.7 mph [6.0 km/h] at 33-ft [10-m] and 6.0 mph [9.7 km/h] at 105-ft [32-m] levels. The design basis wind used for the design and construction of the Lead Cascade facility is 90 mph [145 km/h] at 30 ft [9.0 m] above the ground. No technical basis for the design basis wind is provided in the license application.

Figure 6-1 in the American Society of Civil Engineers Standard—Minimum Design Loads for Buildings and Other Structures (ASCE 7-98) (Ref. 1.3.5.1) identifies a design basis wind speed of 90 mph [145 km/h] for the region. The design basis wind speed used for the design and construction of the proposed American Centrifuge Lead Cascade Facility is consistent with that recommended in ASCE 7-98 (Ref. 1.3.5.1).

Because the proposed American Centrifuge Lead Cascade Facility is not located near the coastal area, hurricanes affecting the coastal area will have no effect on the performance of the facility. Consequently, consideration of hurricane hazards on the design of the proposed American Centrifuge Lead Cascade Facility is not needed.

Based on review of the information about high winds, the staff concludes that high wind hazards and the associated design basis straight-line winds have been addressed acceptably.

iii) Temperature Extremes

Information about the temperature at the PORTS reservation area, where the proposed American Centrifuge Lead Cascade facility is located, is provided in Sections 1.3.3.1, "Regional Climatology," and 1.3.3.3, "Local Meteorology," of the license application (Ref. 1.3.5.22) and Section 1.3, "Meteorology," of the redacted Integrated Safety Analysis Summary (Ref. 1.3.5.23).

The climatological summary shown in Table 1.3-1 of the Redacted Integrated Safety Analysis Summary (Ref. 1.3.5.23) indicated that temperature extremes are events potentially applicable to the facility. The applicant provided the monthly average and extreme temperatures published by the National Oceanic and Atmospheric Administration from the National Weather Service station at the Greater Cincinnati Airport for various measurement periods. The observed temperature extremes for the Greater Cincinnati Airport area from 1962 to 1988 range from -25 to 103 ^{0}F [-32 to 39 ^{0}C].

There is a meteorological tower at the PORTS reservation area. The temperature data measured at the site from 1998 through 2001 was provided by the applicant (Ref. 1.3.5.17), and it was determined that the temperature extremes at the site are within the range of the temperature extremes measured for the Greater Cincinnati Airport area.

The staff reviewed the temperature information and found the information acceptable because recognized data sources were used.

iv) Extreme Precipitation

Sections 1.3.3.1, "Regional Climatology," and 1.3.4.3.1, "Effects of Local Intense Precipitation," of the license application (Ref. 1.3.5.22); and Sections 1.3, "Meteorology," and 1.4.3.1, "Effects of Local Intense Precipitation," of the redacted Integrated Safety Analysis Summary (Ref. 1.3.5.23) discussed the precipitation at the proposed American Centrifuge Lead Cascade Facility site. Table 1.3-2 of the license application listed the precipitation as a function of recurrence interval for various durations. These data were from the National Weather Service for durations from 30 minutes to 24 hours, except for the precipitations for the 10,000 years return period. The precipitations for the 10,000 years return period were extrapolated based on the data from 1 to 100 years and a nonlinear least-squares method.

The staff reviewed the information presented in the license application and the redacted Integrated Safety Analysis Summary for the extreme precipitation and found the information acceptable because recognized data sources, such as the National Weather Service, were used. The least-squares method used to predict precipitations for a return period of 10,000 years is acceptable to the staff.

v) Snow

Sections 1.3.3.1, "Regional Climatology," 1.3.3.3, "Local Meteorology," and 1.3.4.3.1, "Effects of Local Intense Precipitation," of the license application (Ref. 1.3.5.22) and Section 1.3, "Meteorology," of the redacted Integrated Safety Analysis Summary (Ref. 1.3.5.23) briefly discussed the regional and local snowfall. The average annual and the maximum monthly snowfalls recorded at the proposed American Centrifuge Lead Cascade facility site were provided.

According to Figure 7-1 of American Society of Civil Engineers (ASCE) 7-98 (Ref. 1.3.5.1), a 2-percent annual probability of the ground snow load being exceeded (i.e., a 50-yr mean recurrence interval) at the proposed American Centrifuge Lead Cascade Facility site is approximately 20 psf [0.96 kPa]. This value was used by the applicant (Ref. 1.3.5.22) as the design basis ground snow load for the proposed American Centrifuge Lead Cascade Facility. The design basis maximum ground snow load used for the design of the proposed American Centrifuge Lead Cascade Facility is acceptable to the staff because that value is consistent with the value suggested by the ASCE 7-98 standard.

vi) Lightning

Section 1.3.3.3, "Local Meteorology," of the license application (Ref. 1.3.5.22) described the potential of lightning strikes at the proposed American Centrifuge Lead Cascade Facility site. The Applicant indicated that the proposed American Centrifuge Lead Cascade Facility site has less than 40 days of thunderstorms per year that may produce lightning strikes. The applicant points out further in the license application that the proposed American Centrifuge Lead Cascade Facility and the associated power systems are designed and built with heavy grounding or lightning protection to handle lightning strikes.

The design approach used by the applicant to protect the proposed American Centrifuge Lead

Cascade Facility from lightning effects is acceptable.

D) Hydrology

Section 1.3.4 of the application describes the surface hydrology on and around the PORTS site. The PORTS reservation is located near the southern end of the Scioto River basin, which has a drainage area of 6,517 square miles. The headwaters of the Scioto River form in Auglaize County in north central Ohio. The river flows 235 miles through nine counties in Ohio, and through the cities of Columbus, Circleville, Chillicothe, and Portsmouth. At Portsmouth, in Scioto County, the river empties into the Ohio River at river mile (RM) 35B.5. The slope of the Scioto River channel averages about 1.7 ft/mile between Columbus and Portsmouth. The river flows measured at Higby, Ohio, from 1930 to 1991 range from 177,000 cubic feet per second (cfs) on January 23, 1937, to 244 cfs on October 23, 1930, and average 4,654 cfs. The 1937 flood had a peak water elevation of 593.7 ft.

Water used at the site normally comes from groundwater. Currently, all water is supplied by wells in the Scioto River alluvium. These wells are located near the east bank of the Scioto River, downstream from Piketon. Four well fields have the capacity to reliably supply between 36.4 and 40.2 cfs.

The plant nominal elevation is 670 ft, which is about 130 ft above the normal stage of the Scioto River. Both groundwater and surface water at the site are drained from the plant site by a network of tributaries of the Scioto River. The top-of-slab floor elevations for the Lead Cascade facilities are at approximately 671 ft. Storm water that falls at the site is drained to local Scioto River tributaries by storm sewers. The flow of storm water is further controlled by a series of holding ponds downstream from the storm sewer outfalls. The perimeter road, as shown in Figure 1.3-6 (located in Appendix A of the license application), serves as a hydrologic boundary that prevents storm water runoff from backing up into the Lead Cascade facility. The stage of the 1937 flood was 593.7 ft above mean sea level (amsl).

To assess whether failures of the local dams could conceivably jeopardize the safety of systems, the applicant considered holding ponds, lagoons, and retention basins formed by these dams in the local drainage analysis. The surface elevations of all but the X-611B lagoon are well below the 670-ft minimum grade elevation of the Lead Cascade facilities. The water elevation of the X-611B sludge lagoon at 668.8 ft is close to the 670-ft minimum grade elevation at the Lead Cascade facility. The elevation of the top of the dam forming the lagoon is 676.3 ft and exceeds the 670-ft minimum. However, when the conservative estimate of flood wave height (4/9 of the dam height) is used, the flood elevation resulting from a break in the dam would be only 652.8 ft.

The nominal, top-of-grade elevation at the site is 670 ft, about 99 ft above the probably maximum flood (PMF) plus wind wave activity flood stage of 571 ft. The top-of-slab floor elevation for the Lead Cascade facility is at approximately 671 ft. The Scioto River during a PMF superimposed with wind wave activity; therefore, would not inundate these buildings.

The PORTS reservation water supply facility near the Scioto River, pump house X-608, and groundwater well fields, may expect flooding. Though the well fields are designated to operate

during floods, the impacts of flooding on the Lead Cascade cooling system would not result in a release of UF_6. Closing strategic valves can isolate the enrichment process, and during severe conditions all or part of the cascade can be shut down.

E) Geology

i) Seismic Hazards

Seismic hazard is discussed in Section 1.3.6, "Geology and Seismology," of the license application (Ref. 1.3.5.22) and Section 1.5, "Geology and Seismology," of the redacted Integrated Safety Analysis Summary (Ref. 1.3.5.23).

The seismic design criteria for the proposed American Centrifuge Lead Cascade Facility are derived from the seismic hazard criteria developed for the Portsmouth Gaseous Diffusion Plant in the early 1980s and most recently summarized in Revision 1 of the plant's Safety Analysis Report (Ref. 1.3.5.24). In that Safety Analysis Report, seismic design criteria were based on the 1,000-yr return period ground motions, as derived from probabilistic seismic hazard analysis results (Ref. 1.3.5.21 and 1.3.5.3). Hence, the seismic design of the Lead Cascade facility also is based on the 1,000-yr return period seismic ground motions, as derived in the early 1980s.

Staff determined that the 1,000-yr criterion, in concert with appropriate administrative controls and the potentially low off-site consequences from the release of any license material in the event of an earthquake, is sufficient to ensure that the performance requirements of 10 CFR 70.61 are met. Therefore, evaluation of the seismic hazard for the proposed American Centrifuge Lead Cascade Facility in this Safety Evaluation Report is based on adequacy of the geologic and seismologic information used to establish the 1,000-yr return period ground motions.

The following areas concerning the seismic hazard applicable to the safety analysis and design of the proposed facility were reviewed:

1. Seismic Source Characterization;

2. Ground Motion Attenuation;

3. Seismic Hazard Calculation;

4. Development of Site-Specific Spectra; and

5. Surface Faulting.

<u>Seismic Source Characterization</u>

Geological and Tectonic Settings: The license application provides a description of the local and regional geological and tectonic settings. The license application notes that the PORTS reservation is located within the Interior Low Plateaus physiographic province, bordered on the north and west by the Central Lowlands physiographic province and on the south and east by the

Appalachian Plateau physiographic province. Bedrock beneath the PORTS reservation consists of relatively flat-lying and unfaulted carbonate and clastic strata of Paleozoic age. The region also contains unconsolidated Quaternary lacustrian deposits related to Pleistocene glaciation of Eastern North America and pre-glacial alluvial and fluvial deposits related to the ancient Portsmouth River.

The Interior Low Plateaus physiographic province lies within the stable craton of the North American tectonic plate and just to the west of the Appalachian orogenic belt. Active tectonic deformation of the Appalachians ended in the Permian or early Triassic (approximately 240 million years ago), as the orogen was rifted open in response to the break up of Pangea and formation of the Atlantic Ocean basin. The low levels of earthquake activity in the region, including the Appalachians, are generally considered to be associated with preexisting zones of weakness in the crust that formed in the distant geologic past. These zones of weakness are characterized by deeply buried and poorly characterized faults, some of which accomplish a periodic release of strain that builds up continually in the North American continental plate. At the PORTS reservation, postulated earthquakes that could impact safe operation of the proposed facility are associated with zones of crustal weakness in western Ohio, the Appalachians, and the New Madrid seismic zone.

Historical Seismicity: The license application provides a brief summary of historical seismicity at the site. The license application notes that no historical earthquakes have occurred within a 25-mi [40-km] radius of the site. Within 50 mi [80 km] of the site, the largest historical earthquake had an epicenter intensity of IV on the Modified Mercalli scale, which is roughly equivalent to a peak ground acceleration of approximately 0.02 g at the site.

A summary of the state historic seismicity compiled by the Ohio Department of Natural Resources, Division of the Geological Survey[1] indicates low levels of historic seismicity at the PORTS reservation site. It was reported also that more earthquakes have occurred in western Ohio than in other areas of the state. At least 40 felt earthquakes have occurred in western Ohio since approximately 1875, although most of these earthquakes caused little or no damage. The exceptions were two earthquakes in 1937, March 2 and March 9, which caused some damage in Anna, Ohio (e.g., toppled chimneys, cracked plaster, broken windows, and structural damage to buildings).

Northeastern Ohio has experienced approximately 20 felt earthquakes since 1836. Most of these events were small and caused little or no damage. However, one earthquake with a body wave magnitude (m_b) of 5.1 struck on January 31, 1986. This earthquake occurred in northeastern Ohio and caused minor to moderate damage, including broken windows and cracked plaster, in the epicentral area located within Lake and Geauga Counties. Southeastern Ohio has been the site of less than 10 felt historic earthquakes. Earthquakes in 1901 near Portsmouth (Scioto County), in 1926 near Pomeroy (Meigs County), and in 1952 near Crooksville (Perry County), Ohio, caused minor to moderate damage. The license application notes that the peak ground motion recorded at the PORTS reservation was 0.005 g, in 1955.

[1] http://www.dnr.state.oh.us/geosurvey/geo_fact/geo_f03.htm

The most recent probabilistic seismic hazard study of the site was conducted by Risk Engineering, Inc., (Ref. 1.3.5.19). The Risk Engineering, Inc., study relies directly on seismic source characterizations provided by the Electric Power Research Institute (ref. 1.3.5.9) and Lawrence Livermore National Laboratory (Ref. 1.3.5.16) seismic hazard studies for the Eastern United States. Earthquake source characteristics associated with the seismic zones and historic seismicity discussed previously are consistent with information used in both the Electric Power Research Institute and Lawrence Livermore National Laboratory. The seismic hazard spectra developed from the Electric Power Research Institute and Lawrence Livermore National Laboratory studies form the basis for the seismic analysis used by the applicant for design of the proposed Lead Cascade facility.

NRC staff previously accepted the Lawrence Livermore National Laboratory and Electric Power Research Institute data, seismic sources, seismic hazard methods, and results (Ref. 1.3.5.17) for use at sites in the Central and Eastern United States. Thus, the staff concludes that applying the Lawrence Livermore National Laboratory and Electric Power Research Institute hazard results are technically sound. The information about seismic source characterization presented in the license application provides a safety assessment of potential hazards due to natural phenomena on the Lead Cascade facility. The NRC staff finds this information acceptable and in compliance with the safety requirements in 10 CFR 70.65(b)(1).

Ground Motion Attenuation

Seismic hazards used to define the uniform hazard spectra at the PORTS reservation are based on the Lawrence Livermore National Laboratory and Electric Power Research Institute probabilistic seismic hazard studies. Details of ground motion attenuation functions used in the analyses and the relative weights given each model for computing the hazard are described in detail in Risk Engineering, Inc. (Ref. 1.3.5.19). In addition, both the Lawrence Livermore National Laboratory and Electric Power Research Institute probabilistic seismic hazard studies provide methods and results to develop site amplification factors that convert the hard rock uniform hazard spectra to soil hazard spectra. Soil profiles were chosen for till-like shallow soils that are typical for the Central Lowlands physiographic province.

Ground motion attenuation models used in the Lawrence Livermore National Laboratory and Electric Power Research Institute studies provide representative and accurate models of ground motion attenuation characteristics in the Central and Eastern United States. Both studies have captured diverse opinions in the scientific community. NRC staff previously accepted the Lawrence Livermore National Laboratory and Electric Power Research Institute ground motion modeling for sites in the Central and Eastern United States (Ref. 1.3.5.15). Application of these models to the PORTS reservation hazard assessment, and consequently, to the proposed Lead Cascade facility design is considered technically sound. As such, the information about ground motion attenuation presented in the license application is acceptable because it demonstrates compliance with regulatory requirements in 10 CFR 70.65(b)(1).

Seismic Hazards Calculations

The Lawrence Livermore National Laboratory and Electric Power Research Institute results for probabilistic seismic hazard were combined according to the methodologies described in the

DOE Standard (Ref. 1.3.5.7) to obtain uniform hazard spectra for the PORTS reservation site. In the combination, the Lawrence Livermore National Laboratory and Electric Power Research Institute results were given equal weight to obtain an overall representation of the seismic hazard and its associated uncertainty. Details of the process are described in Risk Engineering, Inc., (Ref. 1.3.5.19). Resulting mean hazard and fractile curves are presented for both soil and rock site conditions as well as the median uniform hazard spectra for the 2.0×10^{-3}, 1.0×10^{-3}, and 2.0×10^{-4} exceedance probabilities. The mean peak ground accelerations (soil site) corresponding to the 2.0×10^{-3}, 1.0×10^{-3}, and 2.0×10^{-4} exceedance probabilities from the combined Lawrence Livermore National Laboratory and Electric Power Research Institute results were estimated to be 0.10 g, 0.15 g, and 0.19 g.

The methodology used to combine the Lawrence Livermore National Laboratory and Electric Power Research Institute studies is acceptable. The choice of equal weight is justified because both studies constitute expert elicitation that incorporates a diverse set of scientific results and opinion. The NRC staff previously accepted the Lawrence Livermore National Laboratory and Electric Power Research Institute data, seismic sources, seismic hazard methods, and results (Ref. 1.3.5.15) for sites in the Central and Eastern United States. Thus, the staff concludes that using the combined Lawrence Livermore National Laboratory and Electric Power Research Institute hazard results, including the 1,000-yr return period 0.15-g peak ground acceleration is technically sound. As such, the information about ground motion attenuation presented in the license application is acceptable because it demonstrates compliance with regulatory requirements in 10 CFR 70.65(b)(1).

Development of Site-Specific Spectra

Site-specific design spectra were determined following the procedures in the DOE Standard (Ref. 1.3.5.7) and described in detail in Risk Engineering, Inc. (Ref. 1.3.5.19). In particular, the procedures involve deaggregating the hazard to obtain the two dominant magnitudes and distance pairs that control the peak ground acceleration and maximum spectral velocity. Deterministic response spectral shapes associated with these two magnitude-distance pairs are then calculated and scaled. A single spectrum is created that envelopes the two calculated response spectra. In a study from The Risk Engineering, Inc. (Ref. 1.3.5.19), the rock conditions hazard was used to obtain the response spectra for hard rock conditions. The resulting design spectra were then transformed to soil conditions using a site-specific, soil-response analysis (Ref. 1.3.5.20).

The Sykora and Davis (Ref. 1.3.5.20) study used soil column data, including seismic velocities from 15 soil columns investigated in the late 1970s (Refs. 1.3.5.6 and 1.3.5.12). These site-specific studies revealed that soils at the PORTS reservation site consist of 29.5 ft [9 m] Pleistocene-age lacustrian deposits overlying alluvium, then shale and sandstone. Incorporation of the site-specific soil data led to important refinements in the seismic hazard and design spectra at the site, especially in the shape of the soil response spectra. In particular, the new spectra increased high-frequency ground motions but significantly decreased the ground motions with spectral frequencies below approximately 5 Hertz (Hz). Ground motions at low spectral frequencies (less than ~ 2 Hz) were as much as 80 percent lower than those used in the original PORTS design. The implication of these results is that for spectral frequencies below 5 Hz, the effective exceedance probability is much lower than 1.0×10^{-3}, perhaps by as much as one order

of magnitude lower. Thus, the revisions to the seismic hazard information at the PORTS reservation site have shown that the seismic design of those systems, structures, and components with critical frequencies below 5 Hz is conservative.

The methodology and results used to develop site-specific spectra are acceptable. These methodologies follow modern practice. Thus, the staff concludes that the site-specific hazard and response spectra are technically sound. As such, the information presented in the license application is acceptable because it demonstrates compliance with regulatory requirements in 10 CFR 70.65(b)(1).

Surface Faulting

There is no geologic, geophysical, or seismological evidence of active surface faulting at or nearby the PORTS reservation site. Available data also suggest that the Paleozoic bedrock beneath the site is unfaulted. Therefore, surface faulting is not considered a credible disruptive event for the proposed American Gas Centrifuge Lead Cascade Facility.

ii) Slope Stability

Slope stability is discussed in Section 1.3.6.4, "Engineering Geology," of the license application (Ref. 1.3.5.22) and Section 1.5.1.6, "Liquefaction Potential," of the redacted Integrated Safety Analysis Summary (Ref. 1.3.5.23).

As indicated in the license application, the slopes at the proposed American Centrifuge Lead Cascade Facility have a horizontal-to-vertical ratio of less than three. These low-inclination slopes have a static safety factor of greater than 2.0 and a dynamic safety factor greater than 1.5, for a peak ground acceleration of 0.21 gravity.

In review of the safety factors reported in the license application for the slopes at the site, the staff concludes that these slopes pose no threat of instability. The staff site visit further confirmed that the area at the proposed American Centrifuge Lead Cascade Facility is relatively flat. Consequently, slope stability is not a safety concern for this proposed facility.

iii) Liquefaction

Liquefaction potential of soils beneath the proposed American Centrifuge Lead Cascade Facility is discussed in Sections 1.3.6.4, "Engineering Geology," and 1.3.6.7, "Liquefaction Potential," of the license application (Ref. 1.3.5.22) and Section 1.5.1.6, "Liquefaction Potential," of the redacted Integrated Safety Analysis Summary (Ref. 1.3.5.23). the applicant states in the license application that most soils at the proposed American Centrifuge Lead Cascade Facility site are cohesive and exhibit a low potential for liquefaction.

An extensive geotechnical investigation was conducted in the surrounding area, including the proposed American Centrifuge Lead Cascade Facility site, by the Law Engineering Testing Company (Ref. 1.3.5.12). The investigation results show that the soils beneath the proposed American Centrifuge Lead Cascade Facility site consist of mainly inorganic silts and clays with a plastic limit of 50 percent or less. These soils contain 28-percent fines content (smaller than

#200 sieve) or more with an average of 43 percent. The thickness of the soil layers range from approximately 30 to 40 ft [9 to 12 m], and the groundwater level is approximately 10 to 15 ft [3 to 4.6 m] below the ground surface. The standard penetration test results indicate that the majority of the soil samples tested beneath the proposed American Centrifuge Lead Cascade Facility site have a blow count of more than 20. The smallest blow count obtained is 8 and represents only one sample.

Based on characteristics of the soils at the site, Law Engineering Testing Company (Ref. 1.3.5.12) and the applicant (Ref. 1.3.5.22) concluded that the soils at the site have little potential for liquefaction.

The liquefaction potential for soils also can be estimated using an approach suggested by the National Center for Earthquake Engineering Research (Ref. 1.3.5.13). This approach presented cyclic resistance ratio curves for various fines contents in a cyclic stress ratio (CSR)-blow count diagram (Figure 1). These curves are developed based on field observations for earthquakes with Richter magnitudes near 7.5.

For earthquakes of different magnitudes, the cyclic resistance ratio (CRR) curves shown in Figure 1 must be scaled up or down on the diagram. The suggested magnitude scaling factors can be found in the Summary Report for the Proceedings of the National Center for Earthquake Engineering Research Workshop on Evaluation of Liquefaction Resistance of Soils (Ref. 1.3.5.13). According to the license application (Ref. 1.3.5.22) for the proposed American Centrifuge Lead Cascade Facility site, the design basis earthquake has a magnitude of 5.25 with a maximum peak horizontal ground acceleration of 0.15 g. The magnitude scaling factor for this earthquake magnitude is at least 1.43. The magnitude scaling factor of 1.43 is suggested for an earthquake of magnitude 5.5 and may be considered as a bounding value compared with the values suggested by other researchers (Table 3) (Ref. 1.3.5.13). Applying this scaling factor, the CRR curves in Figure 1 represent a safety factor of at least 1.43 for earthquakes of magnitude 5.25.

The CSR can be calculated using the following equation (Ref. 1.3.5.13)

$$CSR = 0.65\left(\frac{a_{\max}}{g}\right)\left(\frac{\sigma_t}{\sigma_e}\right)r_d \qquad (1)$$

where a_{max} is the peak horizontal acceleration at ground surface, g is the acceleration of gravity, σ_t is the total vertical overburden stress, σ_e is the effective vertical overburden stress, and r_d is a stress reduction factor. The stress reduction factor r_d is overburden depth-dependent and can be estimated roughly (Ref. 1.3.5.13) by

$$\begin{aligned} r_d &= 1.0 - 0.00765z \qquad for\ z \le 9.15\ m \\ r_d &= 1.174 - 0.0267z \qquad for\ 9.15\ m \le z \le 23\ m \end{aligned} \qquad (2)$$

where z is the overburden depth in meters.

Using Eqs. (1) and (2) and assuming the groundwater level is 10 ft [3 m] below ground, for soils with a density of 1.76×10^3 kg/m^3 [110 pcf unit weight] (Ref. 1.3.5.12), the staff determined that the CSRs for soils at overburden depths of 20 ft, 30 ft, and 40 ft [6.1 m, 9.1 m, and 12.2 m] are approximately 0.13, 0.146, and 0.144, respectively. The blower counts from a majority of standard penetration test results is more than 20; thus, the aforementioned CSRs fall in a region on the right side of the CRR curves indicating there is no liquefaction potential for the soils at the site. Considering the smallest blow count value, 8, experienced for one of the soil sample tests, the aforementioned CSRs still will fall within the no liquefaction region defined by the CRR curve with 35 percent fines content.

The staff reviewed the geotechnical investigation report, conducted an independent assessment, and concludes that liquefaction of soils at the site is not a safety concern for the proposed American Centrifuge Lead Cascade Facility.

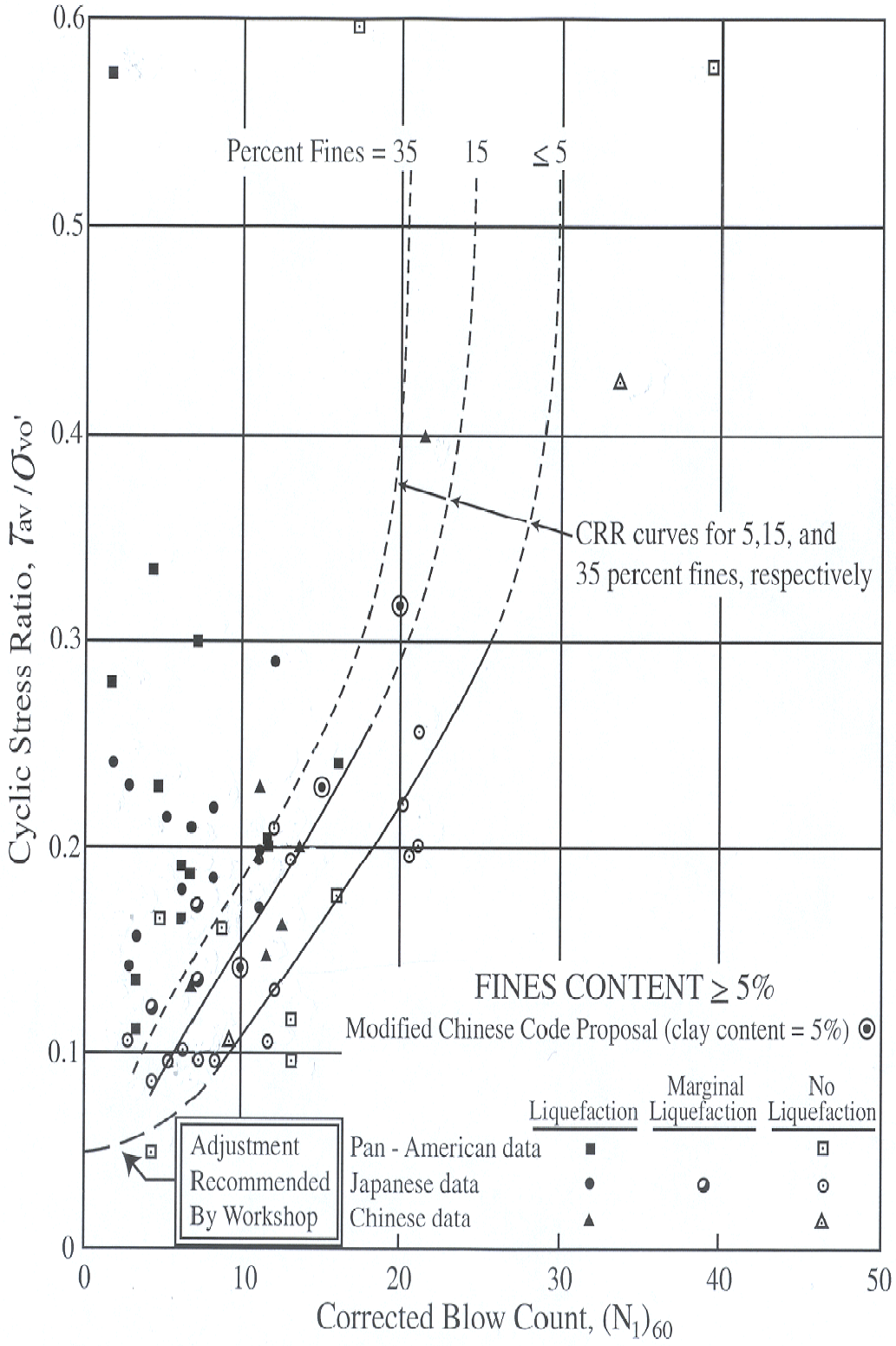

Figure 1. Simplified base curve recommended for calculation of CRR from the standard penetration test results (modified) (Ref. 1.3.5.13)

iv) Settlement

Settlement of foundations for the proposed American Centrifuge Lead Cascade Facility is discussed briefly in Section 1.3.6.4, "Engineering Geology," of the license application (Ref. 1.3.5.22) and Section 1.5.1.6, "Liquefaction Potential," of the redacted Integrated Safety Analysis Summary (Ref. 1.3.5.23). In its license application, the applicant states that the predicted total settlement of foundations is less than 2 in [5.1 cm].

The foundations for the process building were designed and constructed with individual spread footings and piers to support building columns (Ref. 1.3.5.8). Differential settlements between footings could result because of soil variability. In its geotechnical investigation report, Law Engineering Testing Company (Ref. 1.3.5.12) provides the average anticipated settlement of footings for the X-3001 Process Building based on the soil property data determined for the site (Figure 10-7A) (Ref. 1.3.5.12). The geotechnical investigation report also estimated a ±25 percent variation in soil properties within a 50-ft [15.2-m] distance.

Assuming a column load of 400 kips [181×10^3 kg] and a bearing pressure of 6,000 psf [287.3 kPa], the settlement of the footing can be estimated to be approximately 0.76 in [1.93 cm] (Figure 10-7A) (Ref. 1.3.5.12). Thus, a differential settlement of 0.38 in [0.97 cm] {50 percent of the settlement, 0.76 in [1.93 cm], of a footing} can be estimated within a 50-ft [15.2-m] distance.

A separate settlement calculation was performed by Pro2Serve (Ref. 1.3.5.18) as the request of the applicant. The result shows an approximate 0.5 in [1.27 cm] of footing settlement and a differential settlement of 0.25 in [0.64 cm] within a 50-ft [15.2-m] distance, assuming a column load of 400 kips [181×10^3 kg] and a bearing pressure of 6,000 psf [287.3 kPa]. This estimated settlement value is 0.26 in [0.66 cm] smaller than that estimated from the Law Engineering Testing Company site investigation report. The difference may be largely because of the difference in the soil property data set used. The settlement estimate suggested by Pro2Serve uses the data of the soils located in the immediate vicinity of the process building, whereas Law Engineering Testing Company uses the soil data collected for a wider region including those soils below the process building. The methods used for settlement calculations also may be responsible for the difference. The Law Engineering Testing Company used the Westergaard stress distribution method (Ref. 1.3.5.25). Pro2Serve used two methods: the Westergaard stress distribution method to determine the stress increase with depth beneath the footing, and the Bowles method (Ref. 1.3.5.4) for settlement calculation. The estimate from Law Engineering Testing Company is relatively more conservative for the process building.

By conservatively assuming a ±50 percent variation in soil properties within a 100-ft [30.5-m] distance, the differential settlement between footings is approximately 0.76 in [1.93 cm] based on Law Engineering Testing Company and 0.5 in [1.27 cm] based on Pro2Serve.

The distance of two adjacent columns for the process building is 104 ft [31.7 m] along the east-west direction and 20 ft [6.1 m] along the north-south direction. The design basis differential settlement used for the design and construction of the process building is 1 in [2.54 cm] (e.g., Fluor Engineers & Contractors, Inc.) (Refs. 1.3.5.10 and 1.3.5.11) irrespective of the distance between two adjacent footings. This design basis differential settlement is larger

than those estimated by Law Engineering Testing Company and Pro2Serve for a column load of 400 kips [181×10^3 kg] and a bearing pressure of 6,000 psf [287.3 kPa]. Pro2Serve (Ref. 1.3.5.18) also calculated the footing settlement using the actual footing size from the as-built structural drawings and the actual design footing loads. The estimated footing settlement and differential settlement within a 100-ft [30.5-m] distance is approximately 0.3 in [0.76 cm]. This value is substantially smaller than the design basis value.

The staff reviewed the information presented for calculating differential settlements for the process building and finds that the differential settlements determined between adjacent columns of the process building are based on site-specific soil data, and the methodologies used are acceptable. The staff also finds that the design basis differential settlement used for the design and construction of the process building is acceptable because it bounds the estimated differential settlements.

1.3.4 EVALUATION FINDINGS

The staff reviewed the information presented in the license application and in the redacted Integrated Safety Analysis Summary and found reasonable assurance that this information satisfies the requirements of 10 CFR 70.65(b)(1). The assessment of tornado hazards and the development of design basis tornado at the proposed American Centrifuge Lead Cascade Facility site have been discussed adequately so safety of the site can be assessed and design criteria related to tornado effects can be developed. The hazards associated with high winds, including hurricanes, at the proposed American Centrifuge Lead Cascade Facility site have been adequately determined so safety of the site can be assessed and design criteria for regarding the effect of high winds can be developed.

The staff reviewed the information regarding temperature histories and found that the data used to identify temperature extremes were from reliable sources. The staff concluded that design basis temperature extremes selected by the applicant are acceptable and satisfy the requirements of 10 CFR 70.65(b)(1).

The staff reviewed the information presented in the license application and in the redacted Integrated Safety Analysis Summary regarding extreme precipitation and found that the data were from reliable sources. The assessment of precipitation hazards at the proposed American Centrifuge Lead Cascade Facility site has been discussed adequately so safety of the site can be assessed.

The staff reviewed the information presented in the license application and in the redacted Integrated Safety Analysis Summary regarding the ground snow load at the site and found that the discussions on snow at the proposed American Centrifuge Lead Cascade Facility site are acceptable. The staff also found that the design basis maximum ground snow load for the proposed American Centrifuge Lead Cascade Facility site is consistent with ASCE 7-98 (Ref. 1.3.5.1) and is acceptable.

The staff reviewed the information presented in the license application regarding lightning and found that the discussion provided and the assessment on the effects of lightning strikes were

sufficient and satisfy the requirements of 10 CFR 70.65(b)(1).

The staff reviewed the seismic source characterization, ground motion attenuation, seismic hazard calculations, site response and design ground motions, and surface faulting at the proposed Lead Cascade facility site and concluded that the information was appropriate. The information provided and the seismic design criteria proposed adequately demonstrated, with reasonable assurance, that the proposed Lead Cascade facility can be designed to safely account for potential earthquake hazards. Therefore, staff concluded that the information presented in the license application is acceptable because it complies with regulatory requirements in 10 CFR 70.65(b)(1).

The staff reviewed the information presented in the license application and in the redacted Integrated Safety Analysis Summary and found reasonable assurance that this information satisfies the requirements of 10 CFR 70.65(b)(1) in that slope stability is not a safety concern of the site and need not be considered in the design of the structures relied on for safety for the American Centrifuge Lead Cascade Facility.

The staff reviewed the information presented in the license application and the redacted Integrated Safety Analysis Summary and found reasonable assurance that they satisfies the requirements of 10 CFR 70.65(b)(1) in that assessment of liquefaction potential at the proposed American Centrifuge Lead Cascade Facility site is not a safety concern of the site and need not be considered in the design of the structures relied on for safety.

The staff reviewed the information presented in the license application and in the redacted Integrated Safety Analysis Summary and found reasonable assurance that this information satisfies the requirements of 10 CFR 70.65(b)(1) in that differential settlements of foundations for the proposed American Centrifuge Lead Cascade Facility have been characterized adequately so safety of the site can be assessed. The staff also found that the design basis differential settlement for the footings of the proposed American Centrifuge Lead Cascade Facility site is consistent with that estimated and is acceptable.

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2.0 ORGANIZATION AND ADMINISTRATION

The purpose of the review of the applicant's organization and administration is to ensure that the proposed management policies will provide reasonable assurance that the licensee plans, implements, and controls site activities in a manner that ensures the safety of workers, the public, and the environment. The review also ensures that the applicant has identified and provided adequate qualification descriptions for key management positions.

2.1 REGULATORY REQUIREMENTS

10 CFR Part 70.22, 70.23, and 70.62(d) require a management system and administrative procedures for the effective implementation of health, safety, and environment (HS&E) protection functions concerning the applicant's corporate organization, qualifications of the staff, and adequacy of the proposed equipment, facilities, and procedures to provide adequate safety for workers, the public, and the environment.

2.2 REGULATORY ACCEPTANCE CRITERIA

The acceptance criteria applicable to the Nuclear Regulatory Commission's (NRC's) review of the organization and administration section of the application are contained in Section 2.4.3 of the Standard Review Plan for Fuel Cycle Facilities (NUREG-1520).

2.3 STAFF REVIEW AND ANALYSIS

In Section 2.0 of the application, United States Enrichment Corporation (the applicant) Inc. commits to the following policy:

> *The applicant is responsible for safe operation of the Lead Cascade and is committed to conducting operations in a manner that protects the health and safety of workers and the public, protects the environment, provides for the common defense and security, and is in compliance with applicable local, state, and federal laws and regulations.*

In Section 2.0 of the application, the applicant further states that the operations organization is responsible for the safe operation of the Lead Cascade. Programs and staff organizations are established for the environmental, health, safety, safeguards, security, and quality areas and are provided with sufficient resources to support safe operation of the Lead Cascade. Resources from the applicant at the Portsmouth Gaseous Diffusion Plant (PORTS) are utilized in a number of these programmatic areas to provide day-to-day functional support. Intercompany arrangements are in place with the applicant to provide the necessary support.

To the extent that the applicant relies on existing programs and resources from PORTS, such programs and resources must meet 10 *Code of Federal Regulations* Part 70 requirements.

In Section 2.1 of the application, the applicant describes the organizational commitments, relationships, responsibilities, and authorities for the overall management system to assure the

protection of the health and safety of the workers and the public, protection of the environment, and provide for the common defense and security. This section includes the qualifications, functions, responsibilities, and authorities of the positions in the organizations assigned functions related to environmental, health, safety, safeguards, security, and quality during all stages of the project, from design through refurbishment, start-up, and operation.

In Section 2.2 of the application, the applicant describes the management controls for maintaining the environmental, health, safety, safeguards, and quality programs and the administrative systems to control relationships and interfaces between programs.

In Section 2.3 of the application, the applicant describes its plans and the management controls for pre-operational testing and initial start-up of the Lead Cascade.

Figure 2.1-1 of the application shows the Lead Cascade organization for the design/refurbishment/start-up phase of the facility. Prior to beginning operations, this organization transitions to the one shown in Figure 2.1-2.

The Lead Cascade Manager provides overall direction and management of Lead Cascade operations, and oversees activities to ensure safe and reliable operations. The Regulatory Manager, Engineering Manager, Project Support Manager, Operations and Maintenance Manager, Quality Services Manager, and Training Manager report to the Lead Cascade Manager and manage the activities in their area of responsibility.

Personnel minimum qualifications, functions and responsibilities for key staff positions are described below. Alternates are designated in writing in accordance with procedural requirements to fulfill the responsibilities and authorities of these personnel during their absence from the facility.

According to the application, equivalent technical experience means the substitution of two years of nuclear industry experience for each year of college up to a total of three years. Additionally, 30 semester hours or 45-quarter hours from an accredited college or university may be substituted for the remaining one year of baccalaureate education. Individuals who do not possess the formal educational requirements specified in this section or do not meet the equivalent technical experience defined above are not automatically eliminated where other factors provide sufficient demonstration of their abilities to fulfill the duties of a specific position. These other factors must clearly demonstrate proficiency in the technical area for which the position will be responsible, for example, a license or certification, documented completion of relevant training, or previous experience in the same position at another facility. These factors are evaluated on a case-by-case basis, documented, and approved by the Lead Cascade Manager.

A) Organizational Responsibilities and Qualifications

Section 2.1 lists the responsibilities and qualifications of the Executive Vice President and Chief Operating Officer. The Executive Vice President and Chief Operating Officer has overall responsibility for safe operation of the Lead Cascade and have shutdown and stop work authority

for all portions of the Lead Cascade facility. The Executive Vice President and Chief Operating Officer has as a minimum a bachelor's degree in engineering or the physical sciences or equivalent technical experience, six years nuclear experience, and ten years of management experience (which may be concurrent with the nuclear experience).

The Senior Vice President, who reports to the Executive Vice President and Chief Operating Officer, is responsible for the Quality Assurance Program and for determining the status, adequacy, and effectiveness of the Quality Assurance Program Description (QAPD). The Senior Vice President has, as a minimum, a bachelor's degree in engineering or the physical sciences or equivalent technical experience, six years nuclear experience, and 10 years management experience (which may be concurrent with the nuclear experience).

The Director, Enrichment Technology, who reports to the Senior Vice President, has responsibility for program management of Advanced Enrichment Projects including process development, engineering, business and strategic planning, and deployment management. The Director, Enrichment Technology has, as a minimum, a bachelor's degree in engineering or the physical sciences or equivalent technical experience, and six years of experience in engineering, production, and project management, in the nuclear industry.

The Lead Cascade Manager, who reports to the Director, Enrichment Technology, is responsible for the day-to-day safe operation of the facility, for compliance with all applicable NRC regulatory requirements, and for adherence to applicable policies. The Lead Cascade Manager has, as a minimum a bachelor's degree in engineering or the physical sciences or equivalent technical experience, six years of nuclear experience, and six years of management experience (which may be concurrent with the nuclear experience).

The Regulatory Manager, who reports to the Lead Cascade Manager, is responsible for oversight functions in environmental, health, and safety areas. The Regulatory Manager is also responsible for the safeguards and security programs for the Lead Cascade. The Regulatory Manager has as a minimum, a bachelor's degree in engineering or the physical sciences or equivalent technical experience, and four years of nuclear experience.

The Radiation Protection Manager (RPM) is responsible for the Radiation Protection (RP) Program for the Lead Cascade and for the PORTS. The RPM has direct access to the Lead Cascade Manager and the Director, Enrichment Technology for radiation protection matters, and has stop work authority for activities not being conducted in accordance with radiation protection requirements and policies. The RPM has, as a minimum, a bachelor's degree in engineering, health physics, radiation protection, or the physical sciences or equivalent technical experience, and four years experience in radiation protection, including six months at a uranium processing facility.

The Operations and Maintenance Manager, who reports to the Lead Cascade Manager, is responsible for the assembly of centrifuge machines, transportation of machines to and from the cascade, and operation of equipment in the Lead Cascade facility. The Operations and Maintenance Manager has, as a minimum, a bachelor's degree in engineering or the physical sciences or equivalent technical experience, and four years of nuclear experience.

Operations and maintenance supervisors titled Lead Cascade Shift Supervisors and Lead Cascade Maintenance Supervisors, respectively, report to the Operations and Maintenance Manager. Operations and maintenance supervisors have, as a minimum, a high school diploma or satisfactory completion of the General Educational Development test, and three years of industrial/chemical/nuclear plant operations, maintenance, or engineering experience. Supervisors must have one-year supervisory experience or completion of a supervisory training course.

The Engineering Manager, who reports to the Lead Cascade Manager, is responsible for engineering activities in support of operations, including procurement engineering, configuration management, projects (design, fabrication, and construction of facility modifications or additions), system engineering, and business management. The Engineering Manager is responsible for the Nuclear Criticality Safety (NCS) program and maintaining the Integrated Safety Analysis (ISA) and ISA Summary for the facility. The Engineering Manager has, as a minimum, a bachelor's degree in engineering or the physical sciences or equivalent technical experience, and four years of nuclear experience.

The Training Manager, who reports to the Lead Cascade Manager, is responsible for preparation, presentation, and documentation of employee orientations, and for technical and qualification training program development and implementation. During the refurbishment/startup phase, the Training Manager is also responsible for the development and implementation of the procedures program. The Training Manager has, as a minimum, a bachelor's degree in engineering or the physical sciences or equivalent technical experience, and four years of nuclear experience.

The Project Support Manager, who reports to the Lead Cascade Manager, is responsible for the Fire Safety, Emergency Management, and Records Management and Document Control programs for the Lead Cascade. The Project Support Manager has, as a minimum, a bachelor's degree in engineering or the physical sciences or equivalent technical experience, and four years of nuclear experience.

The Nuclear Safety and Quality (NS&Q) Manager, who reports to the Vice President, Operations, has the responsibility to exercise oversight of Lead Cascade and PORTS operations to ensure that the health and safety of the public and workers are adequately protected, to ensure compliance with safety, safeguards, and quality requirements and to ensure implementation of policies, procedures and management expectations. The NS&Q Manager also manages the nuclear material control and accountability group. During the refurbishment/start-up phase of the Lead Cascade, the NS&Q Manager provides oversight and guidance to the Quality Services Manager to assure that Quality Assurance policies and procedures are being appropriately implemented for the Lead Cascade. During the operations phase of the Lead Cascade Project, the NS&Q Manager provides independent oversight and assessment to ensure that the health and safety of the public and workers are adequately protected, to ensure compliance with safety, safeguards, and quality requirements and to ensure implementation of policies, procedures, and management expectations. The NS&Q Manger has as a minimum a technical degree and 15 years nuclear experience with three years of management experience in quality assurance, nuclear safety oversight, engineering and technical support, or regulatory affairs.

The Plant Shift Superintendent (PSS), who reports to the PORTS Shift Operations Manager and is also the PSS for PORTS, represents the Lead Cascade Manager and has the authority and responsibility to make decisions as necessary to ensure safe operations, including stopping work and placing the facility in a safe condition. The PSS is responsible for accumulation and dissemination of information regarding facility activities, serving as or designating an incident commander during emergencies, and making notification of events. The PSS provides the Lead Cascade with a centralized point for incident identification, screening, and reporting. The PSS has, as a minimum, a bachelor's degree in engineering or the physical sciences or equivalent technical experience and four years experience at a uranium processing facility, or a high school diploma plus 12 years experience at a uranium processing facility.

The Fire Services Manager, who reports to the PORTS Plant Services Manager, is responsible for the day-to-day operation of Fire Services, including interpretation and application of applicable fire codes and standards, and has stop work authority for activities in the Lead Cascade not being conducted in accordance with applicable fire protection requirements. The Fire Services Manager has, as a minimum a bachelor's degree or equivalent technical experience, four years of fire protection experience, and six months of nuclear experience.

The NCS Manager, who reports to the applicant's Director, Engineering, is responsible for the management of Lead cascade and PORTS NCS functions, including administering the NCS program and conducting assessments of program implementation. The NCS Manager has, as a minimum, a bachelor's degree in engineering or the physical sciences or equivalent technical experience, and four years nuclear experience, including six months at a uranium processing facility where NCS was practiced.

The minimum operating shift crew consists of a Lead Cascade Shift Supervisor, one Lead Cascade Operator, and one Radiation Protection/Industrial Hygiene technician.

B) Management Control

Section 2.2 of the application describes the management systems with associated policies, administrative procedures, and management controls to ensure: the Lead Cascade equipment, facilities and procedures; the staff (including training and qualifications); and the programs provide for the protection of the health and safety of workers and the public, protection of the environment, and for the common defense and security.

Activities that are essential for effective implementation of the environmental, safety, and health functions are documented in approved, written procedures, prepared in compliance with a document control program.

The commitment tracking and corrective action management systems are integrated to prioritize Lead Cascade actions consistent with their safety and safeguards significance. Any person working in the facility may report potentially unsafe conditions or activities by submitting a problem report.

Where safety or safeguards might be adversely impacted by cost or schedule considerations, it is the policy of the applicant, as explained in page 2-14 of the license application, to subordinate cost and schedule considerations to ensure adequate treatment of safety and safeguards in full compliance with applicable regulatory requirements.

The integration of Lead Cascade operations and the various programs and requirements is accomplished through a variety of management practices, including review of performance indicators, review of identified events or conditions, multi-discipline reviews by the Facility Safety Review Committee (FSRC), and plant work permit systems that provide the integration in the field of various health, safety, and environmental program requirements and hazard evaluations. Additionally, oversight of the integration of various program elements is provided by the PORTS NS&Q Organization.

The FSRC performs multi-discipline reviews of day-to-day and proposed Lead Cascade activities to ensure that these activities are and/or will be conducted in a safe manner. The FSRC advises the Lead Cascade Manager on matters related to radiation protection, nuclear safety, chemical safety, fire safety, and environmental protection. The specific membership, qualifications, meeting frequency, quorum, functions, responsibilities, and required records are provided in a facility procedure. Auditing and oversight of FSRC activities is the responsibility of the NS&Q Manager.

Subcommittees may be established by the FSRC chairperson to provide assistance in conducting reviews and assessments as described in the FSRC procedure. The FSRC chairperson approves the subcommittee procedures, membership, and member qualifications. The FSRC maintains the overall responsibility for any required reviews.

C) Pre-Operational Testing and Initial Start-Up

According to Section 2.3 of the application, specific plans have been established to ensure the safe and efficient turnover, testing, and start-up of Lead Cascade centrifuge machines, equipment, and support systems. These plans cover the transition from the refurbishment phase to the operations phase of the Lead Cascade Project.

The overall objectives of the pre-operational test program are to ensure that the Lead Cascade facilities and systems, including the items relied on for safety (IROFS):

1. Have been adequately designed and constructed;

2. Meet contractual, regulatory, and licensing requirements;

3. Do not adversely affect worker or public health and safety; and

4. Can be operated in a dependable manner so as to perform their intended function.

The refurbishment contractor is responsible for completion of as-built drawing verification, purging/flushing, cleaning, hydrostatic or pneumatic testing, system turnover, and initial

calibration of instrumentation in accordance with design and installation specifications. The Engineering Manager is responsible for coordination of turnover and acceptance testing for the Lead Cascade.

Integrated systems testing, as a minimum, includes system or component tests required by the pertinent design codes or Quality Assurance Program Description that were not performed by the refurbishment contractor prior to turnover to the applicant. The integrated systems tests are performed following completion of construction, flushing and hydrostatic or pneumatic testing, system turnover, and initial calibration of required instrumentation. Scheduling of the testing is such that it generally occurs prior to UF_6 introduction. Other pre-operational tests, not required prior to UF_6 introduction, may be performed following introduction of UF_6 to the process system.

The purpose of initial start-up testing is to ensure safe and orderly UF_6 separation and control. Examples of initial start-up tests include the leak testing, evacuation, start-up, and filling of a centrifuge machine.

2.4 EVALUATION FINDINGS

The staff has reviewed the organization and administration for the Lead Cascade according to Chapter 2 of the Standard Review Plan. The applicant has described (1) clear responsibilities and associated resources for the design, construction, and operation of the facility and (2) its plans for managing and operating the project. The staff has reviewed these plans and commitments and concludes that they provide reasonable assurance that an acceptable organization, administrative policies, and sufficient competent resources have been established or are committed, to satisfy the applicant's commitments for the design, construction, and operation of the facility per 10 CFR 70.22, 70.23, and 70.62(d).

3.0 INTEGRATED SAFETY ANALYSIS (ISA) AND ISA SUMMARY

Due to the kind of information contained in this chapter, it will not be publicly available pursuant to 10 CFR 2.790(a)(3).

The review of the ISA Summary is marked as "Official Use Only" under Exemption 3 of the Freedom of Information Act exemptions, and the review also has the potential to damage commercial interests if disseminated to persons who do not need the information to perform official Nuclear Regulatory Commission (NRC) activities. The staff review is outlined in Appendix A.

The staff reviewed the information presented in the general design criteria and master specifications for design, fabrication, and erection of structural steel for the X-3001 Process Building regarding the codes and standards to be used for the design of civil structures and found that the codes and standards are appropriately identified.

The staff reviewed the information presented in the design calculations and analysis documents regarding the design loads and approaches used to develop seismic design loads for the X-3001 Process Building and found these design loads were developed based on site characteristics and the approaches for determining the design loads followed acceptable codes and standards. These design loads satisfy the requirement of 10 CFR 70.64(a)(2).

The staff reviewed the information presented in the design calculations and analysis documents regarding design load combinations and found that the load combinations to be evaluated for the design of the X-3001 Process Building are appropriate and follow appropriate guidance (Refs. A.5.3, A.5.4, A.5.5, and A.5.17). The design load combinations satisfy the requirement of 10 CFR 70.64(a)(2).

The staff reviewed the information presented in the design calculations and analysis documents regarding the design of the X-3001 Process Building and found that the design is acceptable.

The staff reviewed the As-Built Structural Drawings of the X-3001 Process Building and found that the design is correctly translated in the drawings.

4.0 RADIATION PROTECTION

The purpose of this review is to determine whether the applicant's radiation protection program is adequate to protect the radiological health and safety of workers and to comply with the associated regulatory requirements in 10 CFR Parts 19, 20, and 70.

4.1 REGULATORY REQUIREMENTS

A) Commitment to Radiation Protection Program Implementation

Regulations applicable to establishment of a radiation protection program are presented in 10 CFR 20.1101, Subpart B, "Radiation Protection Programs."

B) Commitment to an ALARA Program

Regulations applicable to the ALARA program are presented in 10 CFR 20.1101, Subpart B, "Radiation Protection Programs."

C) Organization and Personnel Qualifications

Regulations applicable to the organization and qualifications of the radiological protection staff are presented in 10 CFR 70.22, "Contents of Applications."

D) Commitment to Written Procedures

The regulations applicable to radiation protection procedures and RWPs are presented in 10 CFR 70.22(a)(8), "Contents of Applications."

E) Training

The following regulations apply to the radiation safety training program:

1. 10 CFR 19.12 "Instructions to workers"

2. 10 CFR 20.2110 "Form of records"

F) Ventilation and Respiratory Protection Programs

Regulations applicable to the ventilation and respiratory protection programs are presented in 10 CFR Part 20, Subpart H, "Respiratory Protection and Controls to Restrict Internal Exposure in Restricted Areas."

G) Radiation Surveys and Monitoring Programs

NRC regulations in 10 CFR Part 20 are applicable to radiation surveys and monitoring programs:

1. Subpart F “Surveys and Monitoring”

2. Subpart C “Occupational Dose Limits”

3. Subpart L “Records”

4. Subpart M “Reports”

H) Additional Program Commitments

The following regulations are applicable to the additional program commitments:

1. Subpart L “Records” of 10 CFR Part 20

2. Subpart M “Reports” of 10 CFR Part 20

3. Section 70.74 “Additional Reporting Requirements”

4.2 REGULATORY ACCEPTANCE CRITERIA

The acceptance criteria for the NRC’s review of the radiation protection program are outlined in Sections 4.4.1.3, 4.4.2.3, 4.4.3.3, 4.4.4.3, 4.4.5.3, 4.4.6.3, 4.4.7.3, and 4.4.8.3 of NUREG-1520 (Ref. 4.5.3).

4.3 STAFF REVIEW AND ANALYSIS

A) Commitment to Radiation Protection Program Implementation

In Sections 4.0 and 4.1 of Reference 4.5.1, the applicant describes the Lead Cascade Radiation Protection (RP) program. The program is based on the Portsmouth Gaseous Diffusion Plant (PORTS) RP program. As part of certifying PORTS under 10 CFR Part 76, the Nuclear Regulatory Commission (NRC) has reviewed and approved the PORTS RP program, which, under the provisions of 10 CFR 76.60(d), complies with the applicable provisions of 10 CFR Part 20, “Standards for Protection against Radiation” (Ref. 4.5.4).

The Radiation Protection Manager (RPM) will revise the RP program, as necessary, and submit these changes to the Lead Cascade Regulatory Manager. The RPM is responsible for providing guidance and direction for establishment and implementation of the RP Program to the Regulatory Manager. The PORTS Health Physics (HP) Group reports to the RPM and provides radiological protection support to the facility.

In Section 2.1.6 of Reference 4.5.1, the applicant states the RPM will have, as a minimum, a bachelor’s degree in engineering, health physics, radiation protection, or the physical sciences or equivalent technical experience, and four years experience in radiation protection including 6 months at a uranium processing facility.

The PORTS HP Group is staffed with suitably trained individuals who provide oversight and control of the technical aspects of the program elements that affect RP. The HP Group is independent of the organizations responsible for production, and has an equivalent reporting level.

The applicant commits to implement a program in accordance with the requirements of NRC Regulatory Guide 8.2, "Guide for Administrative Practices in Radiation Monitoring," and the American National Standard Institute (ANSI) 13.2-1969, "Guide for Administrative Practices in Radiation Monitoring."

The applicant has committed to review the radiation protection program content and implementation annually, in accordance with 10 CFR 20.1101(c). The RPM is responsible for writing an annual report based on the review.

The applicant has committed to a radiation protection program in accordance with the acceptance criteria. Therefore, the applicant's commitment is acceptable to the staff.

B) Commitment to an ALARA Program

In Section 4.2 of Reference 4.5.1, the applicant describes the Lead Cascade program to maintain radiation doses as low as reasonably achievable (ALARA). The program is implemented, in accordance with 10 CFR 20.1101, through written procedures.

The goals of the program are to maintain personnel radiation exposures and the release of radioactive effluents in accordance with the ALARA principle and to ensure that no individual will receive a radiation dose in excess of any regulatory limit. The responsibility for establishing the ALARA policy rests with the Executive Vice President and Chief Operating Officer. The Lead Cascade Manager has the overall responsibility and authority for the ALARA program and the Regulatory Manager is responsible for implementing the ALARA Program.

The applicant has committed to establishing an ALARA Committee. The committee monitors selected RP issues, advises Lead Cascade management on RP concerns, reviews proposed designs, work practices, selected suggestions, and selected projects with regard to contamination control and/or ALARA, and establishes the annual contamination control and exposure goals.

The ALARA Committee consists of persons from various functional disciplines of the Lead Cascade. The committee meets at least annually and as directed by the chairperson (RPM), and a quorum consists of five standing committee members or their alternates.

Matters that have or may have an impact on contamination control and/or ALARA are reviewed by the committee, including evaluation of audits, reports of radiation levels, contamination levels, employee exposures, and effluent releases. Technologies for selected job tasks, work practices and completed tasks related to contamination control or ALARA, radiation protection violations, lessons learned, trends, and environmental monitoring reports are also reviewed. The Regulatory Manager ensures recommendations of the committee are documented and tracked for completion.

The applicant has committed to an ALARA program in accordance with the acceptance criteria. Therefore, the applicant's commitment is acceptable to the staff.

C) Organization and Personnel Qualifications

The RPM directs the RP Program and is responsible for the implementation of the RP Program. The RPM reports to the Lead Cascade Manager.

HP Technicians and their managers assist and guide workers in the radiological aspects of the job, and have the responsibility and authority to stop radiological work or mitigate the effect of an activity if they suspect the initiation or continued performance of a job, evolution, or test will result in the violation of approved RP requirements.

The minimum qualifications of the RPM are listed in Section 4.3, Subsection A, of this safety evaluation report (SER). HP Technicians are trained in accordance with an approved qualification standard and are re-qualified every two years. Their training consists of initial, on-the-job, and continuing training.

The applicant has committed to organize and staff a radiation protection program in accordance with the acceptance criteria. Therefore, the applicant's commitment is acceptable to the staff.

D) Commitment to Written Procedures

In Section 4.4 of Reference 4.5.1, the applicant describes the implementation of the radiation protection program using written procedures which are prepared consistent with the requirements of 10 CFR Part 20 and 10 CFR Part 70. The applicant has established a management controls program in which the procedures are prepared, maintained, and made available to appropriate personnel at the facility.

The applicant has committed to using RWPs for activities in Contamination Areas (CAs), High Contamination Areas (HCAs), Airborne Radioactivity Areas (ARAs), Radiation Areas (RAs), High Radiation Areas (HRAs), and other areas as required by the HP group. Contamination control limits are consistent with the guidance in Reference 4.5.5. Staff finds that the applicant's commitment to procedures is consistent with the guidance in Reference 4.5.6, and is therefore acceptable.

Qualified HP personnel are authorized to approve, issue, update, revise, and close RWPs. RWPs may be issued for any period of time up to one year, may be closed at anytime by HP, and are normally closed upon job completion.

The applicant has committed to prepare written procedures in accordance with the acceptance criteria. Therefore, the applicant's commitment is acceptable to the staff.

E) Training

In Section 4.4 of Reference 4.5.1, the applicant describes the radiation protection training program. The applicant has committed to require that each person who enters controlled areas of the Lead Cascade where radioactive material may be encountered to be trained, to the appropriate level (i.e., visitor, general employee, radiation worker, health physics technician), commensurate with the hazard per 10 CFR Parts 19 and 20.

Note: As used by the applicant in Reference 4.5.1, and in this SER, the term "controlled areas" means an area to which access is "managed to protect individuals from exposure to radiation and/or radioactive material," (10 CFR 835.2). The NRC has defined areas delineated for this purpose as "restricted areas." Therefore, the applicant's proposal for management of "controlled areas" at the Lead Cascade is evaluated against NRC's requirements for restricted areas.

The applicant has incorporated the requirements of 10 CFR 19.12 into the radiation protection training program. In Section 4.5 of the application, the applicant describes a Radiological Worker Training Program that outlines the requirements of 10 CFR 19.11 and 19.12 and the workers' responsibilities under the applicants radiation protection program.

Radiation protection training is required biennially for personnel having unescorted access. The radiation worker training program is reviewed and evaluated every two years.

The applicant has committed to evaluate the effectiveness and adequacy of the training program. This is completed through feedback from the student and feedback from the student's supervisors based on job performance after training is completed. Additional assessment of training is provided through the audit program.

The applicant has committed to train its employees in radiation protection in accordance with the acceptance criteria. Therefore, the staff finds the applicant's training program acceptable.

F) Ventilation and Respiratory Protection Programs

In Section 4.6 of Reference 4.5.1, the applicant describes the Lead Cascade Ventilation and Respiratory Protection Programs. The applicant has committed to using fixed and portable ventilation units when the unprotected worker could potentially exceed 0.8 Derived Air Concentration (DAC)-hours of exposure. The local ventilation units are equipped with high efficiency particulate air (HEPA) filters.

The portable HEPA filter units' differential pressure is checked per operating procedure, and the range is based on manufacturer's recommendations or as specified in the technical design basis. All HEPA filter systems are efficiency tested in accordance with American Society of Mechanical Engineers (ASME) N510-1989, "Testing of Nuclear Air-Treatment Systems."

The applicant is committed to maintaining the average air velocity above 100 fpm through openings in uranium sampling and handling hoods containing readily dispersible uranium, and

this velocity is checked at least annually. Glove boxes that could generate airborne radioactivity will be maintained at a negative differential pressure, if used.

The applicant is committed to establishing a respiratory protection program following the requirements of 29 CFR 1910.134 and 10 CFR Part 20 for use, issuance, training, and qualifications for respiratory protection users. RWPs are used to specify respiratory protection for radiological protection purposes and are considered for activities where an individual may be exposed to soluble uranium that may exceed 0.8 DAC-hours or an intake of 1 mg of soluble uranium during a shift.

Respiratory protection is used during entry into posted ARAs, breach of contaminated systems or components, work in areas or on equipment with removable contamination levels greater than 100 times the levels in Table 4.6-1 of Reference 4.5.1, and work on contaminated surfaces with the potential to generate airborne radioactivity. The RPM may authorize situations where respiratory protection use is not practical due to physical limitations, such as heat stress, or is not in accordance with ALARA. Stay time controls and continuous workplace airborne monitoring is provided in these situations.

The applicant has committed to implement ventilation and respiratory protection programs in accordance with the acceptance criteria. Therefore, the applicant's commitment is acceptable to the staff.

G) Radiation Surveys and Monitoring Programs

In Section 4.7 of Reference 4.5.1, the applicant describes its commitment to radiation surveys and monitoring programs, which are based on the requirements of 10 CFR 20 and ALARA principles.

Surveys performed consist of routine, work support, and material release surveys. Qualified HP personnel perform surveys for the purposes of establishing personnel protection equipment or for posting requirements.

The routine survey program determines workplace radiological conditions, effectiveness of contamination control measures, and proper identification and posting of radiological hazards. Survey areas are categorized and scheduled commensurate with their relative radiological hazard and contamination potential. Survey frequencies are based on area occupancy, potential for spread of contamination, and process knowledge. Routine contamination survey frequencies are listed in Table 4.7-1 of Reference 4.5.1.

The applicant is committed to restricting release of materials, equipment, and other items for unrestricted use if removable surface contamination levels equal or exceed those specified in Table 4.6-1 of Reference 4.5.1. The NRC staff finds this acceptable and in accordance with NRC guidance related to unrestricted release of equipment and material (Ref. 4.5.5).

Large areas with removable contamination levels on accessible surfaces exceeding the levels specified in Table 4.6-1 of Reference 4.5.1 will be posted as a CA or HCA, and actions will be taken to determine the source.

The personnel monitoring program includes an administrative control level (ACL) of 500 mrem per year Total Effective Dose Equivalent (TEDE) per person, an intake limit for soluble uranium of 10 mg per week, personnel dosimeters to measure the external exposure of personnel, analysis of personnel exposure, maintenance of exposure records, and a network of Fixed Nuclear Accident Dosimeters (FNADs) situated in the Lead Cascade area to serve as area monitors. Personnel dosimeters are also evaluated for neutron dose.

The applicant is committed to having personnel requiring radiation exposure monitoring per 10 CFR 20.1502(a) wear beta-gamma sensitive dosimeters. The dosimeters are processed and evaluated by a processor holding current National Voluntary Laboratory Accreditation Program NVLAP accreditation from the National Institute of Standards and Technology (NIST). The dosimeters are exchanged at least quarterly (+/-2 weeks) unless authorized in writing by the RPM. Self-reading or alarming dosimeters are used for entry into HRAs or Very High Radiation Areas.

The RPM will perform an evaluation if an individual exceeds 50 percent of the Administrative Control Level (ACL) during a calendar quarter or the ACL in the calendar year. The evaluation, specified by procedure, will determine cause, assess the exposure, and document the results.

HP personnel review external dosimetry results to determine unusual trends or exposures. If this occurs, the individual is removed from further exposure until the cause is determined and management is advised of any special controls or restrictions as a result.

The site submits personnel monitoring information to the Radiation Exposure Information Reporting System (REIRS) based on the personnel exposure database, in compliance with the requirements of 10 CFR 20.2206.

The applicant has committed to having personnel who have the potential to receive intakes resulting in a Committed Effective Dose Equivalent (CEDE) greater than or equal to 0.1 rem CEDE in a year or intakes of 1 mg of soluble uranium per week participate in the routine bioassay program. A description of the program and analytical methods employed are given in Table 4.7-2 of Reference 4.5.1. Bioassay sample frequencies and administrative action levels are given in Table 4.7-3 of Reference 4.5.1.

HP personnel review urinalysis results to determine unusual trends. In the event that bioassay results exceed administrative control levels or as determined by HP, personnel participate in follow-up bioassay monitoring. When intakes are confirmed or suspected to exceed 1 mg of soluble uranium per week, special bioassay studies are performed and an investigation is conducted.

The applicant has committed to routine air sampling in Lead Cascade areas where airborne radioactivity concentrations may exceed 10 percent of the DAC listed in Table 4.7-4 of

Reference 4.5.1, averaged over 8 hours. Airborne radioactivity posting levels are also listed in Table 4.7-4 of Reference 4.5.1. Investigations and special bioassay sampling are performed when air samples exceed 0.8 DAC-hours.

Low-volume, high-volume, and lapel air samplers are used for job coverage and general area sampling. Samples are routinely allowed to decay for a minimum of three days due to radon and radon daughter products. Air sample flow measurement devices are calibrated at least annually, and lapel samplers are calibrated as described by use procedure.

Deficiencies associated with surveys and monitoring program or results that exceed the Lead Cascade administrative control level are dispositioned in accordance with the Quality Assurance Program and Corrective Action Program, which is outlined in Section 11.6 of Reference 4.5.1.

Radiation protection instrumentation and sensitivities are listed in Table 4.8-2 of Reference 4.5.1. The instruments are selected to measure the types and energies of radiation encountered with gas centrifuge enrichment operations. The RPM is responsible for maintaining adequate quantities of calibrated radiation detection and measurement instruments.

Portable radiation detection and measurement instruments are inspected, maintained, and calibrated at least annually or removed from service; they are calibrated following any maintenance, modification, or repair deemed likely to affect operation before being returned to service; calibration sources and equipment used for dose rate instruments are within five percent of the stated value and have documented traceability links to the NIST, large area uranium slabs sources are certified to 10 percent by NIST, and calibration sources for contamination monitoring equipment are within 20 percent for activity and 10 percent for surface emission rate; and portable HP instruments that are in use but do not have a built in automatic functional test feature are source checked daily prior to noon that day, or prior to using the instrument if not used on a daily basis. Instruments with the automatic functional test feature that are in use are checked once a week.

The applicant has committed to having personnel exiting contamination areas and contamination control zones monitor themselves for contamination after removing their protective clothing and prior to leaving the step-off pad area. Equipment and materials are monitored and decontaminated as necessary prior to removal, or are controlled as radioactive material.

Sealed sources containing more than 100 μCi of beta and/or gamma emitting material or more than 10 μCi of alpha emitting material, other than tritium, with a half-life greater than 30 days and in any form other than gas, are tested for leakage and/or contamination at intervals not to exceed six months. Sealed plutonium sources containing 0.1 μCi or more of plutonium are tested, when in use, at least every three months.

Posting criteria for restricted areas of the Lead Cascade are listed in Table 4.8-1 of Reference 4.5.1. Radiological control is provided by controlling access to Lead Cascade areas where radioactive material may be encountered and by requiring that each person who enters those areas or facilities receive the appropriate level of radiological worker training.

The applicant has committed to implement radiation surveys and monitoring programs in accordance with the acceptance criteria. Therefore, the applicant's commitment is acceptable to the staff.

H) Additional Program Commitments

The applicant has committed to maintain radiation protection program records in the form required by 10 CFR 20.2110 and retain them as required by 10 CFR 20.2101 through 20.2106.

Reports and notifications of RP issues are made as required by 10 CFR Part 20, Subpart M, and 10 CFR 70.74.

The site submits personnel monitoring information to the REIRS based on the personnel exposure database, in compliance with the requirements of 10 CFR 20.2206.

Deficiencies associated with surveys and monitoring program or results that exceed the Lead Cascade administrative control level are dispositioned in accordance with the Quality Assurance Program and Corrective Action Program, which is outlined in Section 11.6 of Reference 4.5.1.

The applicant has requested an exemption from 10 CFR Part 20 requirements related to posting and labeling each container of licensed material. The applicant states that the UF_6 feed, product, and depleted uranium cylinders are readily identifiable due to their size and unique construction, and are not routinely labeled as radioactive material. The applicant also states that UF_6 cylinders are constantly attended by qualified radiological workers during movement. These practices are consistent with those at other fuel cycle facilities and do not pose an undue risk to radiological workers safety. These actions are acceptable to the staff, and therefore, an exemption to the requirements of 10 CFR 20.1904(a) is recommended.

4.4 EVALUATION FINDINGS

The applicant has committed to an acceptable radiation protection program that includes:

1. An effective, documented program to ensure that occupational radiological exposures are ALARA;

2. An organization with adequate qualification requirements for the radiation protection personnel;

3. Approved written radiation protection procedures and RWPs for radiation protection activities;

4. Radiation protection training for all personnel who have access to restricted areas;

5. A program to control airborne concentrations of radioactive materials with engineering controls and respiratory protection;

6. A radiation survey and monitoring program that includes requirements for controlling radiological contamination within the facility and monitoring of external and internal radiation exposures; and

7. Other programs to maintain records, report to the NRC in accordance with 10 CFR Parts 20 and 70, and correct for upsets at the facility.

The NRC staff concludes that the applicant's radiation protection program is adequate and meets the requirements of 10 CFR Parts 19, 20, and 70. Conformance to the license application and license conditions will ensure safe operation.

4.5 REFERENCES

4.5.1 United States Enrichment Corporation, "License Application for the American Centrifuge Lead Cascade Facility at USEC's Facilities in Piketon, Ohio," February 11, 2003, AET 03-0001.

4.5.2 United States Enrichment Corporation, "Safety Analysis Report Upgrade (SARUP) for the Portsmouth Gaseous Diffusion Plant," Piketon, Ohio, July 24, 2002.

4.5.3 Nuclear Regulatory Commission (U.S.) (NRC). NUREG-1520, "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility." NRC: Washington D.C. March 2002.

4.5.4 Nuclear Regulatory Commission (U.S.) (NRC). "Compliance Evaluation Report for USEC Portsmouth Gaseous Diffusion Plant." NRC: Washington D.C. January 1999.

4.5.5 Nuclear Regulatory Commission (U.S.) (NRC). "Guidelines for Decontamination of Facilities and Equipment prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material." NRC: Washington, D.C. April 1993.

5.0 NUCLEAR CRITICALITY SAFETY

The purpose of this review is to determine whether the applicant's nuclear criticality safety (NCS) program is adequate to support safe operation of the facility, as required by 10 CFR Part 70.

The review will help to determine whether (1) the applicant has provided for the appropriate management of the NCS program, (2) the applicant has identified and committed to the responsibilities and authorities of individuals for developing and implementing the NCS program, (3) the facility management measures described in 10 CFR 70.62 have been committed to and will support implementing and maintaining the NCS program, and (4) an adequate NCS program is described that includes identifying and committing to the methodologies and technical practices used to ensure the safe operation of the facility, as required by 10 CFR Part 70.

5.1 REGULATORY REQUIREMENTS

10 CFR 70.22 and 70.65 specify the general and additional content of a license application, respectively. The NRC's review should verify if the information provided by the applicant meet these requirements. In addition, the applicant's NCS review should verify compliance with the regulatory requirements in 10 CFR 70.24, 70.61, 70.62, 70.64, 70.72, and Appendix A of 10 CFR Part 70.

5.2 REGULATORY ACCEPTANCE CRITERIA

The acceptance criteria for the NRC's review of the applicant's NCS program are outlined in Sections 5.4.3.1, 5.4.3.2, 5.4.3.3, and 5.4.3.4 of NUREG-1520 (Ref. 5.5.5).

5.3 STAFF REVIEW AND ANALYSIS

The Lead Cascade has a possession limit of 250 kilograms (kg) uranium hexafluoride (UF_6), which includes a limit of 700 grams (g) enriched ^{235}U. The requirements in 10 CFR 70.24 specify that each licensee authorized to possess special nuclear material in a quantity exceeding 700 g of ^{235}U must provide a monitoring system to detect an accidental criticality and alert personnel that the accident occurred. Since the Lead Cascade has a limit on enriched uranium of 700 g of ^{235}U, a Criticality Accident Alarm System is not required in the Lead Cascade to detect an inadvertent criticality and alert personnel. The staff concurs that the applicant is not required to provide a system to detect an inadvertent criticality.

In accordance with the requirements contained in 10 CFR 70.62, the likelihood and risks of an inadvertent nuclear criticality were evaluated in the Integrated Safety Analysis (ISA) (Ref. 5.5.3). The evaluation considered moderation events, maintenance evolutions, machine upset conditions, and cylinder operations. The ISA concluded that there were no inadvertent nuclear criticality accident scenarios that could be identified for the Lead Cascade, due to the small quantity of ^{235}U that is present in the facility. Thus, the requirements of 10 CFR 70.61 for high and intermediate consequence criticality events were met. The Nuclear Regulatory Commission (NRC) staff has

reviewed the applicant's ISA Summary and agrees that no inadvertent nuclear criticality accident scenarios could be identified for the Lead Cascade, and thus, the first two parts of 70.61 are met. 10 CFR 70.61(d) requires that the risk of nuclear criticality accidents be limited by assuring that under normal and credible abnormal conditions all nuclear processes are subcritical, including use of an approved margin of subcriticality for safety. It also requires that preventive controls and measures must be the primary means of protection against nuclear criticality accidents. Accordingly, the applicant provided the Lead Cascade NCS program. The facility established a threshold of 1 weight (wt.) percent or higher enriched ^{235}U and 100 g or more of ^{235}U for determining when an evaluation for NCS considerations of planned operations must be performed. A 100 g ^{235}U limit was chosen as a threshold above which a Nuclear Criticality Safety Evaluation (NCSE) is required. This mass is a factor of 10 below the minimum critical mass at 10 percent ^{235}U enrichment, regardless of whether the material is non-oily, oily, or heterogeneous for a fully reflected system. Because of this threshold, many of the Lead Cascade NCS program features described by the applicant may not be required to be implemented. In this regard, the NCS program provides the framework for a defense-in-depth philosophy to help ensure the risk of inadvertent criticality is maintained acceptably low. The NRC staff review of this program is described below.

A) Nuclear Criticality Safety Program

The NCS program is implemented by NCS procedures. The NCS procedures address Lead Cascade personnel, NCS responsibilities, adherence to Nuclear Criticality Safety Approval (NCSA) requirements, review and approval of fissile material operations, posting and labeling requirements, response to NCSA violations, and NCS training requirements. The NCS program meets the Baseline Design Criteria (BDC) requirements in 10 CFR 70.64(a)(9) concerning application of the double contingency principle in determining NCS controls and items relied on for safety (IROFS) in the design of new facilities.

The objectives of the NCS program include:

1. Preventing an inadvertent nuclear criticality;

2. Protecting against the occurrence of an identified accident sequence in the ISA Summary that could lead to an inadvertent nuclear criticality;

3. Complying with the NCS performance requirements of 10 CFR 70.61;

4. Establishing and maintaining NCS safety parameters and procedures;

5. Establishing and maintaining NCS safety limits and NCS operating limits for IROFS;

6. Conducting NCS evaluations to assure that under normal and credible abnormal conditions all nuclear processes remain subcritical, and maintain an approved margin of subcriticality for safety;

7. Establishing and maintaining NCS IROFS based on current NCS determinations;

8. Providing training in emergency procedures in response to an inadvertent nuclear criticality;

9. Complying with NCS BDC requirements in 10 CFR 70.64(a);

10. Complying with the NCS ISA Summary requirements in 10 CFR 70.65(b); and

11. Complying with the NCS ISA Summary change process requirements in 10 CFR 70.72.

The NRC staff finds these NCS program objectives to be in accordance with the guidance in the standard review plan and thus, acceptable.

With respect to organization and administration, the applicant provided details of the nuclear criticality safety responsibilities including the NCS staff qualifications. The minimum requirements for a qualified NCS engineer are:

1. Baccalaureate in engineering, mathematics, or related science;

2. Familiarization with NCS by having a minimum of one year experience at a comparable facility;

3. Completion of NCS-related training course and KENO V., a training course or equivalent;

4. Performance of at least four evaluations under the direction of a Senior NCS Engineer;

5. Performance of walk-through inspections under the guidance of a qualified NCS Engineer; and

6. Additional training in the physics of nuclear criticality and in associated nuclear safety practices if the trainee does not have a nuclear engineering or physics background.

The minimum requirements for a qualified Senior NCS Engineer are:

1. Completion of the minimum requirements for a qualified NCS Engineer;

2. Performance of the functions of a qualified NCS Engineer;

3. Completion of one year as a qualified NCS Engineer; and

4. Be approved by the NCS Manager (or equivalent).

The NRC staff found these requirements acceptable based on knowledge of comparable education and experience requirements at other fuel facilities.

With respect to management measures, the applicant described the procedure, posting and labeling, change control, and operation surveillance and assessment procedures. The applicant addressed the specific items in the guidance provided in NUREG-1520, and thus, these measures are acceptable to the NRC staff.

In the section on methodologies and technical practices, in the application (Ref. 5.5.3), the applicant described adherence to the American National Standards Institute/American Nuclear Society (ANSI/ANS) Standards (Ref. 5.5.1), their process for NCS evaluations and approvals, the design philosophy and review process, and technical aspects of their application of NCS. The NRC staff reviewed the material and found that, in general, the methodologies and technical practices are in agreement with the guidance provided in the NUREG-1520. Thus, this is acceptable.

Since the possession limits for the Lead Cascade have a limit on enriched uranium of 700 g of ^{235}U, which is less that a critical mass, the NCS review was not as extensive as would be performed for a larger amount of special nuclear material.

5.4 EVALUATION FINDINGS

The NRC staff has reviewed the NCS program for the Lead Cascade facility. The staff has reasonable assurance that:

1. The applicant will have in place a staff of managers, supervisors, engineers, process operators and other support personnel who are qualified to develop, implement and maintain the NCS program in accordance with the facility organization and administration and management measures;

2. The applicant's conduct of operations will be based on NCS methodologies and NCS technical practices, which ensure that the fissile material will be possessed, stored, and used safely according to the requirements in 10 CFR Part 70;

3. The applicant will have in place an NCS program in accordance with the subcriticality of operations and margin of subcriticality for safety requirements in 10 CFR 70.61 (d) and baseline design criteria requirements in 10 CFR 70.64 (a); and

Based on this review, the staff concludes that the applicant's NCS program meets the requirements of 10 CFR Part 70 and provides reasonable assurance for the protection of public safety, including workers and the environment.

5.5 REFERENCES

5.5.1 American National Standards Institute/American Nuclear Society (ANSI/ANS), ANSI/ANS-8.1-1983, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors," ANS, La Grange Park, IL.

5.5.2 American National Standards Institute/American Nuclear Society (ANSI/ANS), ANSI/ANS-8.19-1996, “Administrative Practices for Nuclear Criticality Safety,” ANS, La Grange Park, IL.

5.5.3 USEC, “Integrated Safety Analysis Summary for American Centrifuge Lead Cascade Facility at USEC’s Facilities in Piketon, Ohio.” Redacted Version. Docket No. 70-7003, May 23, 2003.

5.5.4 Nuclear Regulatory Commission (U.S.) (NRC). “Nuclear Criticality Safety Standards for Fuels and Material Facilities," Regulatory Guide 3.71.

5.5.5 Nuclear Regulatory Commission (U.S.) (NRC). NUREG-1520, “Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility.” NRC: Washington D.C. March 2002.

6.0 CHEMICAL PROCESS SAFETY

Due to the proprietary nature of information contained in this chapter, it will not be publicly available. The information is specifically exempted from disclosure by statute under the Exemption 3 of the Freedom of Information Act exemptions and has the potential to damage commercial interests if disseminated to persons who do not need the information to perform official Nuclear Regulatory Commission (NRC) activities. The staff review is outlined in Appendix B.

The staff has evaluated the application using the criteria in Chapter 6 of NUREG-1520. Based on the review of the license application, the NRC staff has concluded that the applicant has described and assessed accident consequences that can result from the handling, storage, or processing of licensed materials and that can potentially have significant chemical consequences and effects. The applicant has constructed a hazard analysis that identifies and evaluates those chemical process hazards and potential accidents and established safety controls. To ensure that the performance requirements in 10 CFR Part 70 are met, the applicant has provided reasonable assurance that the items relied on for safety (IROFS) are available and reliable to perform their safety-related functions when needed. The staff has reviewed these safety controls and the applicant's plan for managing chemical process safety and finds them adequate.

The staff concludes that the applicant's plan for managing chemical process safety and chemical process safety controls meets the requirements of 10 CFR Part 70, and provides reasonable assurance that the public health and safety, and the environment, will be protected.

7.0 FIRE SAFETY

Due to the kind of information contained in this chapter, it will not be publicly available. The information is specifically exempted from disclosure by statute under the Exemption 3 of the Freedom of Information Act exemptions and has the potential to damage commercial interests if disseminated to persons who do not need the information to perform official Nuclear Regulatory Commission (NRC) activities. The staff review is outlined in Appendix C.

The dominant fire risk to safety and health of workers and the public for the proposed process is a fire that could lead to loss of confinement of UF_6. This includes a fire damaging the building structure that leads to possible challenges of centrifuge machines and piping that provide UF_6 confinement.

United States Enrichment Corporation (the applicant) submittals provide sufficient information in accordance with requirements of 10 CFR 70.65 regarding potential fire hazards, consequences, and required controls for the proposed Lead Cascade processes. The NRC staff determined that the applicant demonstrated compliance with the performance requirements of 10 CFR 70.61 for fire protection related to postulated accident scenarios. The applicant has identified a reasonable set of items relied on for safety (IROFS) and defense-in-depth protection to ensure acceptable risks within the performance requirements of 10 CFR 70.61.

The applicant commits to reasonable administrative and engineered controls to minimize the risk of fires and protect against potential exposure to fires and explosion hazards for safe operation of the centrifuge process in the Lead Cascade facility. The NRC staff concludes that, if the IROFS and defense-in-depth protection discussed in the applicant's Integrated Safety Analysis (ISA) Summary, along with safety basis assumptions as described, and the planned programmatic commitments in the license applications are adequately implemented to achieve their intended safety performances, a reasonable assurance is provided that the environment and the health and safety of workers and the public will be protected from fires.

8.0 EMERGENCY MANAGEMENT

The purpose of the Nuclear Regulatory Commission's (NRC's) review of the applicant's emergency management plan is to determine if the applicant has established, before the start of operations, adequate emergency management facilities and procedures to protect workers, the public, and the environment.

Emergency capability is incorporated into the baseline design criteria (BDC) of 10 CFR Part 70, and is intended to ensure control of licensed material, evacuation of personnel, and availability of emergency facilities.

8.1 REGULATORY REQUIREMENTS

10 CFR 70.22(i)(1)(i) specifies when an applicant is not required to submit an emergency plan to the NRC. If an applicant is required to submit an emergency plan, as described in 10 CFR 70.22 (i)(1)(ii), then 10 CFR 70.22(i)(3) contains the information that must be included in the emergency plan. In addition, 10 CFR 70.64(a)(6) requires applicants to address the control of licensed material, evacuation of personnel, and availability of emergency facilities for the design of new facilities.

8.2 REGULATORY ACCEPTANCE CRITERIA

The acceptance criteria for the NRC's review of the applicant's emergency management program are outlined in Section 8.4.3.1 of NUREG-1520 (Ref. 8.5.2).

8.3 STAFF REVIEW AND ANALYSIS

The applicant utilized the existing Portsmouth Gaseous Diffusion Plant (PORTS) Emergency Plan (hereafter the Plan) and the accompanying Emergency Plan Implementing Procedures (EPIPs) to meet the requirements in 10 CFR 70.22 (i)(1)(ii) and (i)(3) for the Lead Cascade. The NRC staff reviewed the following elements against the Plan (December 2002), along with the information provided by the applicant in the license application (Ref. 8.5.5), specific to the Lead Cascade.

A) Facility Description

Chapter 1 of the Plan describes NRC-regulated activities, the PORTS facility and the site, and the area near the site. The current plan does not address details particular to the Lead Cascade facility. The applicant has identified the vent system as an area currently not addressed in the Plan, and has included the pertinent information in the license application. This information and any other information specific to the Lead Cascade will be incorporated into the revised Plan. The applicant has committed to revising the Plan to encompass the operation of the Lead Cascade.

B) Types of Accidents

In Section 8.1.3.2 of the license application, the applicant states that the cascade feed cylinder is located in the X-3001 Process Building. The feed cylinder provides the principle source for nearly all of the accident scenarios in the Lead Cascade. The most severe accident sequence for the Lead Cascade involves the release of the entire 250 kg inventory of UF_6. This scenario has already been analyzed for the PORTS gaseous diffusion plant and is identified in Chapter 2 of the current Plan. The staff agrees that it is reasonable to utilize the analysis performed for PORTS the gaseous diffusion plant to address the most severe accident at the Lead Cascade.

C) Classification of Accidents

Chapter 3 of the Plan describes the plant's system for classifying emergencies as alerts or site area emergencies, and the action levels required by each. The Plan requires an Alert be declared if an accident had led or could lead to a release to the environment of radioactive or other hazardous material and would not require a response by off-site response organizations. The Plan requires a Site Area Emergency be declared if an incident had led or could lead to a significant release to the environment of radioactive or other hazardous material and could require response by off-site organizations to protect persons off site. The definitions for Alert and Site Area Emergency are acceptable and suitable for the Lead Cascade. The EPIPs specify emergency action levels for classifying and declaring an emergency.

D) Detection of Accidents

The applicant states in Section 8.1.2 of the license application that UF_6 leak detection instrumentation and criticality accident detection instrumentation are not utilized in the Lead Cascade. The Lead Cascade is small in area and manned on a continuous basis while UF_6 is present in the system. Abnormal operating conditions and accidents are identified by human observation, based on installed instrumentation and routine operator rounds of the facility. Given the small quantity of UF_6 present in the cascade, and the fact that a release would be readily revealed by the presence of a white cloud of UO_2F_2, this approach is acceptable to the staff.

E) Mitigation of Consequences

Chapters 5 and 6 of the Plan describe the means for mitigating the consequences of an accident. The applicant states that there are no off-site radiological consequences from the Lead Cascade accident sequence; rather, the consequences are predominately concerned with related chemical consequences. Details regarding the projected doses and toxic substance concentrations are provided in the Integrated Safety Analysis (ISA) Summary, along with measures credited in preventing and mitigating the events. As discussed in Chapter 3 and Appendix A of this Safety Evaluation Report (SER,) staff reviewed the ISA Summary and find that adequate preventive and mitigative controls have been proposed for potential accidents.

F) Assessments of Releases

Sections 5.3 and 6.4 of the Plan provide a suitable description of the methods and equipment for assessing releases of radioactive or hazardous material. Assessment actions during an Alert include increased surveillance of plant instrumentation and visual observation of incident conditions and monitoring event conditions for potential changes in the emergency classification level. Assessment actions during a Site Area Emergency include assessment of on-site and off-site exposures regularly to determine if and when on-site sheltering may be required. Additional activities can include performing continuing emergency assessments for mitigating events and protective actions on-site based on on-scene and field monitoring results, release information, and meteorological conditions for radiological or hazardous material releases. Radiation detection equipment is used on-site for normal and emergency response use and the plant maintains emergency monitoring instrumentation for chemically toxic material releases. Personnel involved in an emergency submit urine samples for analyses if exposure to radioactive material is suspected. Depending on the meteorological conditions at the time of the incident and the location of the emergency, fence-line sampling may be conducted by monitoring team personnel. Staff reviewed the applicant's assessment of potential releases documented in the ISA Summary. The staff's evaluation is provided in Chapter 3 and Appendix A of this SER. For emergency planning, staff find that the methods and procedures for estimating source terms, atmospheric dispersion, and health effects are acceptable because they are in accordance with the methods and procedures described in Reference 8.5.6.

G) Responsibilities

Chapter 4 of the Plan provides an adequate description of the responsibilities of plant personnel during an emergency. If an emergency would occur, the General Manager is authorized to declare an emergency and initiate the appropriate response. The applicant states in Chapter 2 of the license application that day-to-day functional support for carrying out the requirements of the environmental, safety, health, safeguards, and security programs would be provided by the United States Enrichment Corporation (the applicant). Thus, the General Manager of the PORTS facility would be also responsible for handling an emergency at the Lead Cascade. The Plant Shift Superintendent (PSS) assumes dual role as Crisis Manager (CM) and Incident Commander (IC) until the Emergency Operations Center (EOC) is activated, at which time the General Manager or designee assumes the role of CM. This is conducted by procedural checklists and, if possible, face-to-face briefings. The EOC is automatically activated for Alerts and Site Area Emergencies. Once the CM responsibilities have been transferred from the PSS to the General Manager, the PSS maintains responsibility of IC at the incident scene. Chapter 3 of the Plan describes those responsible for notifying the NRC Operations Center immediately after notifying the off-site authorities but not later than one hour after the declaration of an emergency. Chapter 7 of the Plan identifies the department responsible for maintaining and updating the Plan.

The PORTS Emergency Management (EM) organization will maintain and update the Plan and distribute the Plan to groups having responsibilities for response functions. EM may change the Plan without receiving prior NRC approval, provided the change does not decrease the effectiveness of the Plan and the NRC and affected off-site response organizations are provided

with copies of any changes within six months of the change. EM has established procedures to allow off-site response organizations to comment on new or significantly updated plans.

Staff find that the responsibilities described in the Plan meet the acceptance criteria described in Reference 8.5.2, and are therefore acceptable.

H) Notification and Coordination

Chapter 3 of the Plan provides a commitment to notify off-site authorities of an emergency. Section 4.3 of the Plan describes provisions for requesting off-site assistance. Sections 5.6 and 5.7 describe medical transportation and treatment of contaminated workers. In its ISA Summary, the applicant has identified mitigative items relied on for safety (IROFS) which would be relied upon during emergencies at the Lead Cascade.

I) Information to Be Communicated

Section 3.3 of the Plan adequately describes the information to be communicated to off-site response organizations and the NRC during emergency notifications. The information communicated includes plant status conditions, radiological and other hazardous materials release data, recommendations for protective actions for off-site response organizations, and other applicable emergency information.

J) Training

Section 7.2 of the Plan describes the type of training that general plant personnel, emergency response organization and support personnel, other Department of Energy (DOE) reservation personnel, and off-site emergency support organizations receive. Training records are retained to document readiness assurance. Formal training programs have been developed and updated, including the General Employee Training, Emergency Preparedness General Employee Training, Specialized Emergency Plan Training for the Emergency Response Organization, and Off-site Emergency Management Training. Personnel working at the Lead Cascade facility will receive additional training on equipment, Job Hazard Analysis, and emergency egress. Sections 7.2.1, 7.2.2, and 7.2.3 of the Plan describe the topics and general content of training programs used for training the licensee's on-site and off-site emergency response personnel. These training courses are provided on a biennial basis.

K) Shutdown (Recovery and Facility Restoration)

Chapter 9 of the Plan describes the means of restoring the facility to a safe condition after an accident. Recovery and restoration activities are conducted such that exposures are maintained in accordance with the as low as reasonably achievable (ALARA) principle. A recovery organization is established and managed by a Recovery Manager, who has overall responsibility for recovery activities. For the Lead Cascade facility, the Lead Cascade Manager is responsible for ensuring the adequacy and appropriateness of recovery operations.

L) Exercises and Drills

Section 7.3 of the Plan adequately describes the provisions in place to conduct quarterly drills and biennial exercises of the emergency plan. The Emergency Management Manager has overall responsibility for implementing a coordinated program of emergency drills and exercises identified in an EPIP. An exercise scenario manual is prepared for each drill and exercise. These scenarios are varied annually. Drill and exercise controllers and evaluators are trained on the proper conduct of emergency exercises. Off-site response organizations are contacted quarterly to check and update telephone numbers, if necessary. Formal critiques are conducted after each drill and exercise. Critique items that have safety significance, indicate a regulatory violation, or reflect serious deficiencies in plan content or implementation, are identified to the Plant Shift Superintendent and a Problem Report is initiated. Emergency equipment and supplies are kept available, properly stored, and maintained in operable status for emergency response personnel to perform their respective duties and responsibilities. These equipment and supplies are inspected, inventoried, and operationally tested quarterly and after each use. There is emergency equipment and supplies available to replace temporarily damaged equipment. Letters of Agreement between PORTS and off-site support organizations and agencies are reviewed and updated at least every four years.

M) Hazardous Chemicals

Chapter 10 of the Plan states that the facility complies with the Emergency Planning and Community Right to Know Act, 42 U.S.C. § 11001 et seq. The Plan and appropriate EPIPs would be used in case of a hazardous chemical release emergency. There are plant administrative procedures for releases that are not considered emergencies. Any hazardous material spill or release is reported to the Plant Shift Superintendent. Appropriate EPIPs describe actions to be implemented on a hazardous chemical release emergency, from evacuating the area of release or spill to terminating the incident and proceeding to the recovery phase.

N) Responsibilities for Developing and Maintaining the Emergency Program and Its Procedures current

Section 8.1.1 of the license application states that the Plan is maintained and updated by the applicant. The applicant may incorporate changes to the Plan without receiving prior NRC approval, provided those changes do not decrease the effectiveness of the Plan and the NRC and affected off-site response organizations receive copies of any changes to the Plan within six months of the change.

8.4 EVALUATION FINDINGS

The NRC staff has evaluated the information related to emergency management submitted by the applicant in Chapter 8 of the license application for the Lead Cascade and other associated documents. The applicant intends to use the current PORTS Emergency Plan, and perform the necessary updates to include information applicable to the Lead Cascade. Further, the applicant commits to perform an additional review of the Plan and incorporate any necessary additional specific emergency management aspects of the Lead Cascade into the existing Plan, including

adding any hazardous chemicals associated with the Lead Cascade to the existing list of chemicals, with clear indication that the hazardous chemicals pertain to the Lead Cascade. As such, the NRC staff is proposing the following condition:

> "The Portsmouth Gaseous Diffusion Plant Emergency Plan shall be revised to appropriately address the Lead Cascade facility and the Emergency Plan Implementing Procedures encompassing these topics will be implemented at least 120 days prior to the introduction of UF_6 in the Lead Cascade facility."

In accordance with 10 CFR 70.22(i), the applicant commits to maintain and execute an emergency plan for responding to the radiological hazards resulting from a release of radioactive material or hazardous chemicals incident to the processing of licensed material. The NRC staff reviewed the Plan with respect to 10 CFR 70.22(i) and the acceptance criteria in NRC's NUREG-1520. The NRC staff determined that the applicant's Plan is adequate to demonstrate compliance with 10 CFR 70.22(i), in that: (1) the facility is properly configured to limit releases of radioactive materials in the event of an accident; (2) a capability exists for measuring and assessing the significance of accidental releases of radioactive materials; (3) appropriate emergency equipment and procedures are provided on-site to protect workers against radiation and other chemical hazards that might be encountered after an accident; (4) a system has been established to notify federal, state, and local government agencies and to recommend appropriate protective actions to protect members of the public; and (5) necessary recovery actions are established to return the facility to a safe condition after an accident.

The requirements of the Emergency Plan are implemented through approved written procedures. Changes that decrease the effectiveness of the emergency plan may not be made without prior NRC approval. In accordance with 10 CFR 70.32(i), the NRC will be notified of other changes that do not decrease the effectiveness of the emergency plan within six months of making the changes.

8.5 REFERENCES

8.5.1 Code of Federal Regulations, *Title 10, Energy*, Part 70, "Domestic Licensing of Special Nuclear Material."

8.5.2 Nuclear Regulatory Commission (U.S.) (NRC). NUREG-1520, "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility." NRC: Washington, D.C. March 2002.

8.5.3 U.S. Nuclear Regulatory Commission, "Standard Format and Content for Emergency Plans for Fuel Cycle and Materials Facilities," Regulatory Guide 3.67.

8.5.4 Portsmouth Gaseous Diffusion Plant, "Portsmouth Gaseous Diffusion Plant Emergency Plan," Revision 62, December 2002.

8.5.5 United States Enrichment Corporation, “License Application for the American Centrifuge Lead Cascade Facility at USEC’s Facilities in Piketon, Ohio,” February 11, 2003, AET 03-0001.

8.5.6 Nuclear Regulatory Commission (U.S.) (NRC), Washington, D.C. “Nuclear Fuel Cycle Facility Accident Analysis Handbook.” (NUREG/CR-6410). March 1998.

9.0 ENVIRONMENTAL PROTECTION

The purpose of the Nuclear Regulatory Commission's (NRC's) review of the applicant's environmental protection plan is to determine whether the applicant's proposed environmental protection measures are adequate to protect the environment and the health and safety of the public as required by 10 CFR Parts 20, 51, and 70.

9.1 REGULATORY REQUIREMENTS

To be considered acceptable, the applicant must satisfy the following regulatory requirements regarding environmental protection:

1. 10 CFR Part 20 specifies the effluent control and treatment measures necessary to meet the dose limits and dose constraints for members of the public specified in Subparts B, D, and F; the survey requirements of Subpart F; the waste disposal requirements of Subpart K; the records requirements of Subpart L; and the reporting requirements of Subpart M.

2. 10 CFR Part 51 specifies that the applicant must establish effluent and environmental monitoring systems to provide the information required by 10 CFR 51.60(a).

3. 10 CFR Part 51 also specifies that the applicant must submit an environmental report, as required by 10 CFR 51.60(b), or support a categorical exclusion as described in 10 CFR 51.22(c).

4. 10 CFR 70.22(a)(7) specifies that the applicant must demonstrate that proposed facilities and equipment, including measuring and monitoring instruments and devices for the disposal of radioactive effluents and wastes, are adequate to protect the environment and the health and safety of the public.

5. 10 CFR 70.65(b) specifies that an applicant for a facility must provide an Integrated Safety Analysis (ISA) Summary that includes a list of the items relied on for safety (IROFS) established by the applicant and other elements.

6. 10 CFR 70.59 outlines the radiological effluent monitoring reporting requirements for a 10 CFR Part 70 licensee.

9.2 REGULATORY ACCEPTANCE CRITERIA

The acceptance criteria for the NRC's review of the applicant's environmental protection program are outlined in Section 9.4.3.2 of NUREG-1520 (Ref. 9.5.2).

9.3 STAFF REVIEW AND ANALYSIS

A) Radiation Safety

i) ALARA Goals for Air and Liquid Effluent Control

With regard to radiological as low as reasonably achievable (ALARA) goals for air effluent control, the applicant proposes an ALARA goal for all PORTS air effluent, including the Lead Cascade facility, of 5% of the 10 CFR 20.1101 constraint of 10 mrem/year. This is less than the 10 mrem/year goal recommended in NRC Regulatory Guide 8.37 (Ref. 9.5.1), Regulatory Position C.1.2, and is, therefore, acceptable to the staff.

For liquid effluents, the applicant proposes a goal of 10% of the air effluent goal, or 0.05 mrem/year to the most exposed member of the public. This is much less than the 10 mrem/year goal recommended in NRC Regulatory Guide 8.37, Regulatory Position C.1.2, and is, therefore, acceptable to the staff.

ii) Air Effluent Controls to Maintain Public Doses ALARA

The applicant has identified several controls to reduce emissions of radioactivity to the atmosphere. These controls include: (1) activated alumina traps to capture uranium hexafluoride in the purge vacuum and evacuation vacuum systems; (2) a continuous vent sampler that draws samples from the process vent using an isokinetic probe that maintains a real time indication of effluent levels; and (3) engineered local ventilation systems to capture residual uranium during maintenance activity around the centrifuges. These engineered features are expected to maintain air effluent releases to less than 1.0 mCi per week, resulting in total uranium concentrations at the site boundary of less than 2.1×10^{-16} μCi/mL. The staff finds that this is approximately 0.007% of the limits in 10 CFR Part 20, Appendix B, Table 2. These controls ensure that levels remain well below regulatory limits and ALARA air effluent goals, and are therefore acceptable to the staff.

iii) Liquid Effluent Controls to Maintain Public Doses ALARA

The applicant proposes to use the X-6619 Sewage Treatment Plan (STP), without physical modification or adjustment of treatment limits, to treat and release wastewater to the Scioto River. The staff reviewed liquid discharge data from this operating facility and find that maximum anticipated liquid discharge concentrations of 2×10^{-8} μCi/mL are well below the values in 10 CFR Part 20, Appendix B, Table 2, and proposed liquid effluent ALARA goals. Therefore, the use of the X-6619 STP for processing the small increment in wastewater generated by the Lead Cascade facility is acceptable to the staff.

The applicant proposes to use the Gaseous Diffusion Plant (GDP) Recirculating Cooling Water (RCW) system to discharge blow down water from the GCEP RCW system. The GDP RCW system is discharged at National Pollution Discharge Elimination System (NPDES) outfall 004 and contains maximum anticipated liquid discharge concentrations of 2×10^{-9} μCi/mL uranium. The staff reviewed liquid discharge data from NPDES outfall 004 and find that maximum anticipated liquid discharge concentrations of 2×10^{-9} μCi/mL are well below the values in 10 CFR Part 20, Appendix B, Table 2, and proposed liquid effluent ALARA goals. Therefore, the use of the NPDES outfall 004 for processing the small increment in RCW generated by the Lead Cascade facility is acceptable to the staff.

The applicant proposes to collect leaks from the machine cooling water (MCW) system and incidental spills elsewhere in the lead cascade in the Liquid Effluent Collection (LEC) system. Water accumulated in two 550 gallon tanks is sampled and pumped to either the X-6619 or the X-705 Liquid Waste Treatment system. Given the small increment in waste volumes from the MCW and existing compliance at both the X-6619 or X-705 outfalls, this proposal to discharge the MCW system to either of these outfalls is acceptable to the staff.

The applicant expects no changes in storm water runoff associated with installation and operation of the Lead Cascade facility. Storm water runoff in the vicinity of the Lead Cascade is captured in either the X-2230N West Holding Pond (NPDES outfall 012) or the X-2230M Southwest Holding Pond (NPDES outfall 013). Each holding pond contains maximum anticipated liquid discharge concentrations of $1x10^{-8}$ μCi/mL uranium and discharges to the Scioto River.

iv) ALARA Reviews and Reports to Management

To maintain effluents below regulatory limits and ALARA goals, the applicant proposes to implement a program based on use of a Baseline Effluent Quantity (BEQ) for uranium and technetium. The BEQ is defined as the maximum effluent expected under normal operating conditions. For the Lead Cascade facility, the new BEQ values are 0.01 mCi per week for the Lead Cascade Process Vent, 5.1 mCi per week for the X-2230 West Holding Pond, and 3.1 mCi per week for the X-2230M Southwest Holding Pond.

The required action described by the applicant for releases above a BEQ include action by the Lead Cascade Shift Supervisor and Plant Shift Superintendent to notify management and initiate an evaluation of whether additional emission controls would significantly reduce public exposure. This approach is consistent with the guidance in NUREG-1520, and is therefore acceptable to the staff.

Additional action levels above the BEQ have been defined by the applicant. For uranium, effluent levels between 2 and 10 times the BEQ will trigger an investigation into the causes of the higher emissions. For releases above an Environmental Protection Agency (EPA) Reportable Quantity (>0.1 Ci in 24 hours), immediate notification of the Plant Shift Supervisor and control of the source or shutdown is specified. Above 1 Ci per week, corresponding to 2 mrem at the maximally exposed individual off-site, the applicant will close the affected vents until control of emissions is re-established. These levels are incremental, resulting in increasingly more aggressive action before a regulatory limit is exceeded, and are therefore acceptable to the staff.

v) Waste Minimization

The applicant proposes to assign to the Lead Cascade facility Regulatory Manager the responsibility for coordinating waste minimization activities at the Lead Cascade. Proposed waste minimization practices would include, for example, the promotion of the use of non-hazardous materials, the use of material substitution practices, process optimization, in-process closed-loop recycle, and waste segregation. The applicant emphasizes the minimization of

hazardous waste, mixed waste, and low-level radioactive waste generation. These practices are consistent with the guidance and are therefore acceptable to the staff.

B) Effluent and Environmental Monitoring

i) Air Effluent Monitoring

With regard to compliance with NRC standards for public dose in 10 CFR 20.1301, the applicant proposes to calculate total effective dose equivalent (TEDE) to the maximally exposed person. The applicant will use the CAP88 package of computer codes, which implements the Regulatory Guide 1.109 model for downwind dispersion, human intake of airborne radioactivity, ground deposition of airborne radioactivity, uptake of radioactivity in foodstuff, and consumption of contaminated food. This approach is acceptable to the staff.

The applicant has identified two potential sources of radioactivity discharges to air from the Lead Cascade facility process: (1) the Evacuation Vacuum (EV) and Purge Vacuum (PV) systems discharged through the X-3001 process vent; and (2) fugitive emissions from the X-7726 static stand captured by local ventilation systems. The applicant does not expect measurable emissions from the X-7726 static stand as a result of opening centrifuges that have operated on uranium hexafluoride gas.

A continuous vent sampler using alumina media would be used to monitor the EV and PV system vents for uranium hexafluoride. Weekly primary sample traps would be analyzed for uranium-234, uranium-235, uranium-238, and technetium-99. A secondary trap would be replaced quarterly. The applicant does not expect to detect technetium-99 in the Lead Cascade, but all vent samplers at PORTS, including those at the Lead Cascade, will be analyzed for technetium-99. Uranium isotope concentrations would be determined using either alpha spectroscopy or Inductively Coupled Plasma/Mass Spectrometry (ICP/MS), with minimum detectable activity less than 0.2% of the 10 CFR 20, Appendix B, Table 2, values.

A representative sample of air effluent is collected using an isokinetic probe and monitoring of both vent and sampler air flows.

The applicant has described its compliance program for Subpart H of 40 CFR Part 61, National Emission Standards for Hazardous Air Pollutants (NESHAP). These standards are consistent with NRC's 10 CFR 20.1101(d), which form the basis for the applicant's air effluent ALARA program goals described above.

ii) Liquid Effluent Monitoring

With regard to compliance with NRC standards for public dose in 10 CFR 20.1301, the applicant proposes to calculate TEDE to the maximally exposed person downstream of plant discharges on the Scioto River. To calculate TEDE, the applicant proposes to use the Regulatory Guide 1.109 model, with consideration of direct use of contaminated surface water for drinking water, the ingestion of fish from contaminated water, and the use of contaminated water for irrigation. This approach addresses the major pathways for uranium, and is therefore acceptable to the staff.

The applicant has identified five potential sources of radioactivity discharges to surface water from the Lead Cascade facility process, including four NPDES outfalls: (1) X-6619 Sewage Treatment Plant identified as NPDES outfall 003; (2) GDP RCW System identified as NPDES outfall 004; (3) X-2230N West Holding Pond identified as NPDES outfall 012; and (4) X-2230M Southwest Holding Pond identified as NPDES outfall 013.

The applicant proposes to use composite samplers and weekly analysis to measure contaminants in leased outfalls that have continuous flow. Outfalls with intermittent flow would be monitored with grab samplers during periods of outfall flow. The applicant proposes to monitor uranium, gross alpha, gross beta, and technetium beta radioactivity. Uranium radioactivity would be measured by ICP/MS, with minimum detectable concentrations of 0.001 *g per milliliter, which results in a minimal detection concentration (MDC) below 2% of the applicable 10 CFR Part 20, Appendix B, Table 2 values. An MDC of less than 5% is acceptable to the staff.

With respect to permits from other agencies, the applicant is required under Condition II.K of the NPDES Permit to submit a quarterly written report to the Ohio EPA summarizing the radioactive discharges from the permitted outfalls. A copy of this report is submitted to the NRC within 60 days of the end of the calendar quarter.

The applicant proposes to use level gauges to detect unplanned releases to groundwater or soil from the Liquid Effluent Control (LEC) underground tanks located south of the Lead Cascade X-3001 Process Building.

iii) Laboratory Quality Control

The staff has reviewed the Lead Cascade facility Quality Assurance Program Plan and found it acceptable. At the X-710 Analytical Laboratory, the applicant proposes to use dedicated Chain of Custody procedures, National Institute of Standards and Technology (NIST-)traceable standards, matrix spikes, duplicate and replicate samples, check samples, and blind and double blind quality control (QC) samples as quality control practices. In addition, the applicant proposes to continue participation in external control programs.

iv) Safe Storage and Timely Disposition of Waste

The applicant has identified a system of waste packaging, labeling, storage, treatment, shipment and disposal for solid and liquid waste. Waste would be packaged to meet Department of Transportation (DOT) and 10 CFR Part 71 requirements.

With respect to storage, the applicant states that the State of Ohio has provided an exemption to the 90-day limit for any Lead Cascade mixed waste. Where outdoor storage is required, waste with removable contamination would be packaged in containers, wrapped or covered to prevent the release of radioactivity. Mixed aqueous waste that cannot be processed in United States Enrichment Corporation (the applicant) facilities would be stored on site until treatment is available at commercial treatment facilities that are licensed in accordance with 10 CFR Part 61, or applicable NRC Agreement State requirements.

Liquid low level radioactive waste and mixed waste solutions requiring treatment are processed in the X-705 Decontamination Building. Processed wastewater would be released to the X-6619 STP.

Off-site shipments of radioactive wastes are manifested in accordance with 10 CFR 20.2006 requirements and are packaged, labeled, and manifested in accordance with applicable State, DOT, NRC and EPA requirements. Disposal is in compliance with 10 CFR Part 20, Subpart K requirements and records are retained in accordance with 10 CFR 20.2108. Any classified waste would be disposed of in accordance with 10 CFR Part 95 and security program requirements.

v) Environmental Monitoring

The applicant has described the existing PORTS environmental monitoring program in the context of the low levels of radioactive effluents expected from the proposed Lead Cascade facility. On the basis of these low levels, the applicant has not proposed significant changes to the PORTS environmental monitoring program. The applicant proposes to demonstrate compliance with NRC radiation standards at the proposed Lead Cascade facility using effluent monitoring and atmospheric dispersion modeling. Given the low levels of effluent expected from the Lead Cascade facility, and the small increment the Lead Cascade facility would contribute to existing PORTS operations, the staff finds acceptable the applicant's proposal to use effluent monitoring and modeling to demonstrate compliance with the regulations of Parts 20 and 70. Legacy groundwater contamination remains DOE's responsibility, including the PORTS trichloroethylene plumes.

vi) ISA Summary

In Appendix D of the application, the applicant demonstrated that the bounding fire event postulated in the ISA would not result in consequences to the environment above the intermediate consequence threshold of 10 CFR 70.61(c)(3). Therefore, no IROFS are identified for the Lead Cascade facility for reducing the environmental risks of natural phenomena and potential accidents. Staff reviewed this calculation and concurs with the applicant’s conclusion.

9.4 EVALUATION FINDINGS

The applicant has committed to adequate environmental protection measures, including (1) environmental and effluent monitoring and (2) effluent controls to maintain public doses ALARA as part of the radiation protection program. The NRC staff concludes, with reasonable assurance, that the applicant’s conformance to the application is adequate to protect the environment and the health and safety of the public and comply with the regulatory requirements imposed by the Commission in 10 CFR Parts 20, 51, and 70.

9.5 REFERENCES

9.5.1 U.S. Nuclear Regulatory Commission, “ALARA Levels for Effluents from Materials Facilities,” Regulatory Guide 8.37, July 1993.

9.5.2 U.S. Nuclear Regulatory Commission. NUREG-1520, "Standard Review Plan for the Review of a License application for a Fuel Cycle Facility." NRC: Washington D.C. March 2002.

10.0 DECOMMISSIONING

The purpose of the Nuclear Regulatory Commission's (NRC's) review of the applicant's decommissioning plan is to conclude, with reasonable assurance, that the applicant will be able to decommission the facility safely and in accordance with NRC requirements.

At the time of the initial license application and again at license renewal, the applicant/licensee may be required to submit a decommissioning funding plan (DFP). The purpose of the NRC's review of the DFP is to determine whether the applicant has considered decommissioning activities that may be needed in the future, has performed a credible site-specific cost estimate for those activities, and has presented the NRC with financial assurance to cover the cost of those activities in the future. The DFP, therefore, should contain an overview of the proposed decommissioning activities, the methods used to determine the cost estimate, and the financial assurance mechanism. This overview must contain sufficient details to enable the reviewer to determine whether the decommissioning cost estimate is reasonably accurate.

10.1 REGULATORY REQUIREMENTS

The following NRC regulations require planning, financial assurance and recordkeeping for decommissioning, as well as procedures and activities to minimize waste and contamination:

10 CFR 70.22(a)(9) "Decommissioning Funding Plan"

10 CFR 70.25 "Financial Assurance and Recordkeeping for Decommissioning"

10.2 REGULATORY ACCEPTANCE CRITERIA

The applicant's decommissioning plan will be considered acceptable if it meets the regulatory requirements mentioned in Section 10.1 and the acceptance criteria outlined in NUREG-1727 (Ref. 10.5.3).

10.3 STAFF REVIEW AND ANALYSIS

The NRC's staff review of the decommissioning plan focused on United States Enrichment Corporation's (the applicant's) conceptual decommissioning activities for the Lead Cascade, the decommissioning cost estimates, and the financial assurance for decommissioning activities. The applicant identifies activities required for decommissioning, and estimates decommissioning costs. Using the cost data as a basis, the applicant states that it has made financial arrangements to cover the costs required for returning the Lead Cascade facilities to the Department of Energy (DOE) in accordance with the terms of the Lease Agreement. The following subsections contain these decommissioning aspects as described by the applicant, and the NRC staff's assessment of the applicant's proposed decommissioning plan, cost estimate, and funding plan.

A) Conceptual Decontamination and Decommissioning Plan

The applicant states there are two locations that have been identified for the Decontamination Service Area: the south half of the X-3001 Process Building and the X-7726 Centrifuge Assembly Area. A final determination has not been made concerning which option to utilize. However, for purposes of the Decommissioning Funding Plan, the applicant assumes that decontamination activity will occur in the south half of the X-3001 Process Building.

i) Decommissioning Program

The applicant states in Section 10.1 of the license application that the plan for decommissioning of the Lead Cascade is to promptly decontaminate or remove materials from the facilities that are required under the Lease Agreement to return the facilities to DOE. Special design features would be incorporated into the facility as part of decommissioning planning. Features related to radioactive contamination control include characterization of areas as clean areas, potentially contaminated areas (contamination control zones), and contaminated areas, and minimization of non-radioactive process equipment and systems in locations subject to likely contamination. These features serve to minimize the spread of radioactive contamination during operation and simplify the eventual facility decommissioning. Features related to worker exposure and waste volume control include ample access for efficient equipment dismantling and removal of equipment that may be contaminated, connections in the process systems for thorough purging at facility shutdown, design drawings of the facility, and controlled worker access to contaminated areas. These features serve to minimize worker exposure to radiation and radioactive waste volumes during decontamination activities.

The NRC staff agrees that the features provided above would minimize worker exposure and waste volumes during decontamination activities.

ii) Decommissioning Steps

The applicant briefly describes the decommissioning methodology to be employed at the Lead Cascade. In Section 10.2 of the license application, the applicant states that implementation of decommissioning may begin immediately following facility shutdown and that the estimated time for decommissioning is slightly greater than one year. The order of steps involved for facility decommission would be: process system purging, equipment dismantling and removal, decontamination, disposition of Confidential and Secret Restricted Data equipment and material, disposal of wastes, and completion of a final radiation survey. The applicant indicates that a new facility would not be required for decontamination of facility components and structures, since the Decontamination Service Area would be the primary location for decontamination activities. The major items from the facility that are expected to require decontamination are described by the applicant in Table 10.2-1 of the license application. These items are located inside the contamination control zones of the facility. The applicant commits to exercise good housekeeping practices during normal operations to maintain the other areas contamination free.

The NRC staff finds acceptable the applicant's proposal to use the Decontamination Service Area for decontamination activities and the commitment to exercise good housekeeping practices to avoid contamination in areas outside the contamination control zones.

For the first decommissioning step, system cleaning, the applicant commits to remove UF_6 material to the fullest extent possible by normal process operation. This would be followed by evacuation and purging of process systems. This process is estimated to take less than a month.

The next step would be the dismantling process. The applicant describes the dismantling process as simple but labor intensive, and indicates its intention to optimize this process. For optimization, the applicant would consider several facts, described in Section 10.2.3 of the license application. Details of the optimization process would be decided near the end of facility useful life. The applicant estimates that the time frame to accomplish both dismantling and decontamination is approximately one year.

The applicant addresses the decontamination process in Section 10.8 of the license application. The NRC's review of the applicant's decontamination process is explained in Section 10.3, Subsection A, Item VIII, of this safety e valuation report.

For the sale/salvage process, the applicant states that items removed from the facility would be categorized as potentially re-usable equipment, recoverable decontaminated scrap, and wastes, and they would be handled accordingly.

During the disposal process, radioactive waste would be disposed of in licensed low-level radioactive waste disposal facilities. The applicant estimates that approximately 2,600 cubic meters of radioactive waste would be generated during the decommissioning operation. This waste may need to be reduced prior to disposal. A more complete estimate would be provided in the applicant plan for completion of decommissioning. Confidential and Secret Restricted Data components and documents at the facility would be either transferred to the commercial plant or would be dispositioned in accordance with 10 CFR Part 95 and the Lead Cascade Security Program.

The final decommissioning step is a final radiation survey to verify that the facility meets regulatory requirements for decommission and can be returned to DOE in accordance with Lease Agreement requirements. This survey would be compared to an initial radiation survey performed prior to operation to subtract the background radiation of the area. This final radiation survey would be documented in a report, the results would be analyzed, and further action would be taken in the decommissioning process, if required.

The NRC staff has found that the decommissioning steps submitted by the applicant are appropriate for the Lead Cascade.

iii) Management/Organization

The applicant states in Section 10.3 of the license application that management of the decommissioning program would assure proper training and procedures are provided to assure

worker health and safety. The programs would focus on minimizing waste volumes and worker exposure to hazardous or radioactive materials. Qualified contractors assisting with decommissioning would be subject to Lead Cascade security and training requirements and procedural controls. The NRC staff expects to receive a more detailed description of the applicant's plans in the decommissioning plan, to be provided at the time of decommissioning.

iv) Health and Safety

The applicant states in Section 10.4 of the license application that a radiation protection program would identify and control sources of radiation, establish worker protection requirements, and direct the use of survey and monitoring instruments, in accordance with the as low as reasonably achievable (ALARA) principles.

The NRC staff has found acceptable the applicant's approach of using the ALARA principle in the radiation protection program to identify and control sources of radiation, establish worker protection requirements, and direct the use of survey and monitoring instruments.

v) Waste Management

The applicant states in Section 10.5 of the license application that radioactive and hazardous wastes produced during decommissioning would be collected, handled, and disposed of in accordance with regulations applicable to the Lead Cascade at the time of decommissioning. Generally, procedures would be similar to those described for wastes produced during Lead Cascade operation. The applicant commits to ultimately dispose of these wastes in licensed radioactive or hazardous waste disposal facilities located elsewhere. Non-hazardous and non-radioactive wastes would be disposed of consistent with good industrial practice and in accordance with applicable regulations.

The NRC staff finds acceptable the applicant's commitment to collect, handle, and dispose of waste produced during decommissioning, in accordance with regulations applicable to the Lead Cascade.

vi) Security/Nuclear Material Control

The applicant commits in Section 10.6 of the license application to maintain the requirements for physical security and for nuclear material control and accountability during decommissioning in a manner similar to the programs in force during Lead Cascade operation. This includes requirements for control of classified information and classified equipment described in the Lead Cascade Security Program and the requirements for control of nuclear materials in the Fundamental Nuclear Materials Control Plan for the American Centrifuge Lead Cascade Facility. Any necessary revisions to these programs would be provided in the applicant plan for completion of decommissioning, to be provided at the time of decommissioning.

vii) Recordkeeping

The applicant commits in Section 10.7 of the license application to maintain records related to spread of contamination, as-built drawings and modifications of structures and equipment, cost estimate performed for the decommissioning funding plan, and funding method used for assuring funds, in accordance with established records management and document control procedural requirements.

The applicant included adequate information about recordkeeping of spills and other unusual occurrences, as required by 70.25 (g)(1), and on certain areas within and outside the restricted areas of the facility, as specified in 70.25 (g)(3)(i) through (iii). Thus, the NRC staff finds the applicant's recordkeeping commitments acceptable.

viii) Decontamination

The applicant briefly describes in Section 10.8 of the license application the facilities, procedures, and expected results of decontamination for the Lead Cascade. The applicant provides in Table 10.2-1 of the license application a list of components and structures for potential decontamination at decommissioning, including an estimated quantity for each component or structure.

The applicant states in Section 10.8.1 of the license application that the Decontamination Service Area, which would be one of three locations described in Section 10.0 of the license application, would be available for decommissioning. This space is needed for handling centrifuges to be disassembled and dispositioned, along with the UF_6 vacuum pumps, valves, and piping. Unusable material would be destroyed. The specialized area would have a disassembly area, a buffer stock area, a decontamination area, and a scrap storage area. Equipment that may be found in the decontamination area include transport and manipulation equipment, dismantling area, cutting machines, dismantling boxes and tanks, degreasers, citric acid and demineralized water baths, contamination monitors, wet blast cabinets, crushers or size reduction equipment, shredders, and scrubbers. The applicant states that a decontamination facility is not needed during Lead Cascade normal operation.

The NRC staff finds acceptable the applicant's proposed plans for the Decontamination Service Area, which is assumed to be the south half of the X-3001 Process Building for purposes of the DFP. The NRC staff would need to re-evaluate this area if the applicant decides to change the Decontamination Service Area. Staff would expect this change to be included in a revised DFP submitted as committed to in Section 10.10.1 of the applicant's license application.

The applicant briefly describes the procedures for decontamination in Section 10.8.2 of the license application. These procedures would be developed and approved by facility management to minimize worker exposure and waste volumes, and to assure work is carried out in a safe manner. At the end of useful facility life, some of the equipment, most of the buildings, and the outdoor areas should already be acceptable for release for unrestricted use. If they were inadvertently contaminated during Lead Cascade operation, the applicant has stated that they

would likely be cleaned up when the contamination is discovered. This limits the scope of necessary decontamination at the time of decommissioning.

The operations that would be performed to the centrifuges include removal of external fittings; removal of bottom flange, motor and bearings, and collection of contaminated oil; removal of top flange and withdrawal and disassembly of internals; and destruction of classified parts by shredding, crushing, burial, etc.

At the time of decommissioning, the NRC staff will review and evaluate the proposed decontamination activities in the final decommissioning plan. The NRC staff will also review the facility and site characterization data. The NRC staff will authorize decontamination work if it concludes that the facility and site decontamination are consistent with the applicable safety and safeguards regulations.

The applicant states in Section 10.8.3 of the license application that recoverable items would be externally decontaminated and suitable for reuse except for a very small amount of intractably internal contaminated material that severely limits potential customers. Other than centrifuge machines, there is potentially a small amount of salvageable scrap material. Material requiring disposal would primarily be process piping, trash, and residue from the effluent treatment systems. No problems are anticipated which would prevent the Lead Cascade facilities from being released to DOE in accordance with Lease Agreement requirements. In light of the above, the staff concludes that regardless of the requirements of the Lease Agreement, at the time of the decommissioning, the applicant would have to comply with the license termination requirements of Subpart E of 10 CFR Part 2, unless an exemption has been approved by the NRC.

ix) Agreements with Outside Organizations

The applicant states in Section 10.9 of the license application that the decommissioning plan and the funding arrangements described in Section 10.10 of the license application provide for decontamination of the Lead Cascade for turnover to DOE. As such, no agreements with outside organizations are required for control of access to the facility following shutdown and decommissioning.

B) Financial Assurance for Decommissioning

The applicant submitted a DFP in accordance with 10 CFR 70.22(a)(9). This plan describes the cost estimate and funding mechanism for decommissioning. It also includes a section on the adjustment of the cost estimate and funding, and recordkeeping plans related to decommissioning funding.

i) Decommissioning Costs

Section 3.0 of the DFP contains the decommissioning estimate, which includes a table of cost activities and the total estimated cost in 2002 dollars. Section 3.0 refers to Chapter 10 of the license application for more detailed information. Chapter 10 of the license application contains more discussion of the cost activities and how the estimated costs were calculated.

The NRC staff finds that the DFP includes sufficient information and meets the requirements of 10 CFR 70.22 (a)(9) and 10 CFR 70.25 (e). The DFP provided adequate information, in accordance with NUREG-1727, about:

1. Descriptions of facility buildings and grounds (including number and dimensions of areas) that will require decontamination, number and type of components that will require decontamination, potential levels of contamination, and potential quantities of materials or waste requiring disposition;

2. Labor costs and non-labor costs; and

3. Key assumptions used in the decommissioning cost estimate.

The applicant provided adequate information regarding the contingency factor that is applied to the cost estimates to provide reasonable assurance for unforeseen circumstances that could increase decommissioning costs, and the method for and frequency of adjusting the costs in the DFP to account for inflation and other factors. This information included the use of a 25% contingency factor, a description of the method for adjusting the costs in the DFP, and a frequency of adjusting the costs of three years or less.

The NRC staff finds acceptable the applicant's decommissioning cost estimate and that it is in accordance with the regulatory requirements.

ii) Funding Arrangements

The DFP includes a model Standby Trust Agreement and a model Surety Bond as described in NUREG-1727. The format and content of the model Surety Bond and model Standby Trust Agreement are adequate, however, specific information regarding the licensee and the trustee is not included because the documents are not executed originals.

The NRC staff finds that the executed original Surety Bond should include:

1. Name and address of the licensee;

2. Corporate or partnership information, including the state in which it is incorporated in;

3. Name, address, and phone number of the surety and the type of organization and its state of incorporation;

4. Total penal sum of the bond based on the cost estimate in the DFP; and

5. Signature blocks that are consistent with the licensee and surety information.

The NRC staff also finds that the executed original Standby Trust Agreement should include:

1. Name of the licensee;

2. Corporate or partnership information, including the state in which it is incorporated in;

3. Name, address, and phone number of the trustee;

4. Type of financial assurance instrument that is planned to fund the trust;

5. Signature blocks that are consistent with the licensee and trustee information; and

6. Schedules A, B, and C that are consistent with the information provided above, including the amount of the trust fund and fees paid to the trustee.

The NRC staff finds that the applicant's funding arrangements for decommissioning should meet the regulatory requirements in 10 CFR 70.22(a)(9) and 10 CFR 70.25 (e), provided the executed originals of the Surety Bond and the Trust Fund Agreement are reviewed and approved by the NRC when they include the specific information as indicated above. Accordingly, the NRC proposes the following condition:

> "Ninety days before commencing nuclear operations, the applicant shall provide NRC with a certification in the form of an executed original document(s) that financial assurance for decommission has been provided in the amount of the cost estimate for decommissioning, as documented in the Decommissioning Funding Plan."

10.4 EVALUATION FINDINGS

The NRC staff has evaluated the applicant's plans and financial assurance for decommissioning in accordance with NUREG-1727. On the basis of this evaluation, the NRC staff has determined that the applicant's plans for decommissioning (based on the assumption that the south half of the X-3001 Process Building will be used for this purpose) comply entirely with the NRC's regulations and provide reasonable assurance of protection for workers, the public, and the environment. The NRC staff would need to re-evaluate this area if the applicant decides to change the Decontamination Service Area. Staff would expect this change to be included in a revised/updated DFP submitted as committed to in Section 10.10.1 of the license application.

The staff also finds that the applicant's financial assurance for decommissioning requires that executed originals of the Surety Bond and the Standby Trust Agreement must be reviewed and approved as acceptable to the NRC prior to the licensee possessing licensed material.

10.5 REFERENCES

10.5.1 Code of Federal Regulations*, Title 10, Energy*, Part 70, "Domestic Licensing of Special Nuclear Material."

10.5.2 Nuclear Regulatory Commission (U.S.) (NRC). NUREG-1520, "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility." NRC: Washington, D.C. March 2002.

10.5.3 Nuclear Regulatory Commission (U.S.) (NRC). NUREG-1727, "NMSS Decommissioning Standard Review Plan." NRC: Washington, D.C. September 2000.

10.5.4 Nuclear Regulatory Commission (U.S.) (NRC). NUREG-1757, "Consolidated NMSS Decommissioning Guidance." NRC: Washington, D.C. September 2003.

11.0 MANAGEMENT MEASURES

Management measures are functions, performed by the applicant, generally on a continuing basis, that are applied to items relied on for safety (IROFS) to provide reasonable assurance that the IROFS are available and able to perform their functions when needed.

The purpose of this review is to verify whether the applicant provided sufficient information to conclude, with reasonable assurance, that the management measures applied to IROFS, as documented in the Integrated Safety Analysis (ISA) Summary, provide reasonable assurance that the IROFS will be available and able to perform their functions, when needed, consistent with the performance requirements of 10 CFR 70.61. If a graded approach is used, the review should also determine whether the measures are applied to the IROFS commensurate with the IROFS' importance to safety.

11.1 REGULATORY REQUIREMENTS

The requirements for fuel cycle facility management measures are specified in 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material."

1. 10 CFR 70.4 states that management measures include configuration management (CM), maintenance, training and qualifications, procedures, audits and assessments, incident investigations, records management, and other quality assurance (QA) elements.

2. 10 CFR 70.62(a)(3) states that records must be kept for all IROFS failures, describes required data to be reported, and sets time requirements for updating the records.

3. 10 CFR 70.62(d) requires an applicant to establish management measures, for application to engineered and administrative controls and control systems that are identified as IROFS, pursuant to 10 CFR 70.61(e), so they are available and reliable to perform their functions when needed.

4. 10 CFR 19.12 states requirements for workers instructions that are applicable to personnel training and qualifications.

5. 10 CFR 70.22(a)(8) states requirements for procedures in license applications.

6. 10 CFR 70.72 requires a licensee to establish a CM program.

7. 10 CFR 70.74 (a) and (b) states requirements for incident investigation and reporting.

11.2 REGULATORY ACCEPTANCE CRITERIA

The acceptance criteria for the Nuclear Regulatory Commission's (NRC's) review of the applicant's management measures program are outlined in Section 11.4.3 of NUREG-1520 (Ref. 11.5.1).

11.3 STAFF REVIEW AND ANALYSIS

A) Configuration Management Program

i) CM Policy

The goal of the CM program should be to ensure that the facility has accurate, current documentation that matches the facility's physical/functional configuration, while complying with applicable requirements.

The applicant has defined the Configuration Management Policy for the Lead Cascade and provided a program overview in Section 11.1 of the license application. The CM program will include: (1) identification and documentation of IROFS, including delineation of IROFS' boundaries; (2) organizational descriptions of duties and responsibilities; and (3) administrative controls, procedures, and policies to implement and document activities that maintain the facility's configuration.

Program Overview

The Engineering Manager has primary responsibility for the implementation of the CM program. The CM program will be applied to the facility, structures, processes, systems, equipment, components, computer programs, and activities of personnel, regardless of the item's Quality Level (QL) classification. The CM program will be applied in a graded approach, taking into account the ten items listed in Item III, "QA Program," in Section 11.3, Subsection H, "Other QA Elements," of this Safety Evaluation Report (SER) Chapter:

The QLs are established as described in Item III, "QA Program," in Section 11.3, Subsection H, "Other QA Elements," of this SER Chapter.

Implementing procedures will be established to provide a management system to evaluate, implement, and track each change.

Key Program Responsibilities

The Engineering Manager has been assigned primary responsibility for implementation of the CM program. Key program responsibilities have been identified and address the engineering, procurement and materials, operations, maintenance, records management and document control, procedures, training, and quality functions during the refurbishment/start-up and operational phases.

ii) Design Requirements

The applicant states that design requirements for IROFS or other systems or components required to support the ISA and meet the performance requirements of 10 CFR 70.61 are developed in accordance with 10 CFR 70.64. Support of environmental impact and mission-based functions are identified in System Requirements Documents (SRDs). Design requirements

are developed by the engineering organization and documented in Design Criteria Documents (DCDs). The Engineering Manager is the Facility Design Authority and approves the DCDs. The baseline configuration of the facility consists of the approved DCDs and ISA Summary, Design Basis Documents, SRDs, and the as-built drawings and specifications. All changes to these documents are controlled.

iii) Document Control

The applicant stated that the procedure control program described in Section 11.4 of the license application assures that procedures are generated, reviewed, approved, and distributed in a controlled manner. An index of documents required to be controlled to support the CM program will be maintained. The Records Management and Document Control Program are described in Section 11.7 of the license application.

iv) Change Control

The applicant stated that changes to the physical facility (IROFS and supporting systems and components), procedures, and controlled documents will be implemented in accordance with 10 CFR 70.72. The Facility Safety Review Committee will review changes to the facility or facility's operations, including tests and experiments, as specified in procedures, and will also specify approval authority for changes. The change control process will be implemented in accordance with approved procedures that cover:

1. Requests for engineering assistance for acceptability review, engineering approval and disposition;

2. 10 CFR 70.72 reviews of facility modifications, additions, or changes to determine the need for prior NRC approval;

3. Evaluation of permanent and temporary modifications for any required changes or additions to the facility's procedures, personnel training, testing programs, or the ISA Summary;

4. Identification of critical repair parts for IROFS; and

5. Independent, technical reviews of proposed changes.

A final review prior to release to operations will be conducted to verify:

1. The safety analysis documentation is complete and approved;

2. Operational procedure changes, if required, are completed and other supporting procedure changes have been initiated;

3. Operational training and qualification changes, if required, have been completed;

4. Design changes have been completed and any as-built changes are identified and approved including review for adequacy;

5. Document changes, if required, are completed;

6. Temporary change duration is documented and the modified equipment is tagged;

7. Post-modification testing has been successfully completed; and

8. All appropriate approvals have been obtained.

Changes to procedures and controlled documents will be controlled in accordance with the programs described in Sections 11.4, "Procedures," and 11.7, "Records Management," of the license application.

v) Assessments

The CM Assessment Program includes both initial and periodic assessments that include document assessments and system walk-downs. An initial assessment will be performed prior to operation of the Lead Cascade to provide for field verification of design requirements and documentation and verification of procedures and training. Deficiencies or recommendations will be documented and addressed in accordance with the corrective action program described in Section 11.6 of the license application.

vi) Design Reconstitution

The applicant will lease portions of structures to house the Lead Cascade that were already built by the U.S. Department of Energy for the Gas Centrifuge Enrichment Program. The applicant intends to use existing facilities and equipment to fill many of the design requirements. To verify that the design of the existing facilities and equipment meet the Lead Cascade Design Requirements, the applicant will utilize a verification process that will include:

1. Preparation of Assessments Plans for existing structures, systems, and components (SSCs);

2. System walk-downs;

3. Issuance of Assessment Report to identify any discrepancies for the as-built configuration;

4. Comparison of the configuration documented in the Assessment Report to the SRD;

5. Evaluation and verification that the existing SSCs meet the SRD requirements;

6. For any necessary changes, development of design criteria and design change packages;

7. Documentation and specifications are updated after construction; and

8. Design criteria and documents are incorporated into the Lead Cascade Baseline Configuration.

B) Maintenance

The maintenance function is described in Section 11.2 of the license application. Policies, procedures and programs will address personnel qualifications and training, design/work control, corrective and preventive maintenance, surveillance and monitoring, post-maintenance testing, control of measuring and test equipment, and equipment/work history. Organizational responsibilities are defined.

Qualification and Training

Personnel qualification and training requirements for maintenance personnel will be established that take into consideration the importance of the activities performed under each job classification to the safe operation of the facility and maintenance of IROFS, the complexity of the activity, the frequency of the task, and the consequences of performance errors. Consideration is also given to skill-of-the-craft and availability of procedures. Contractors performing work on IROFS will be given the same guidance under the oversight of the Lead Cascade organization.

Design/Work Control

The applicant has stated that a work control process will be established to provide for the control, review, and approval process to maintain the documented configuration of Lead Cascade systems. The level of maintenance planning, the extent of reviews, and the approval required to perform the maintenance task will be graded based on the QL of the item. Prior to removing an IROFS from service, appropriate compensatory measures will be established. The repair and replacement of IROFS will be performed with like-for-like parts. Modifications will only be performed following evaluation and approval of the engineering organization. Approval of both the maintenance organization and equipment owner are required prior to the performance of maintenance on equipment. The operations organization will be required to authorize the performance of maintenance and removal of IROFS from service. The overall work control process provides configuration control of Lead Cascade equipment.

i) Corrective Maintenance

The corrective maintenance process will address those actions to check, troubleshoot, and repair equipment that has degraded or failed. Corrective maintenance will be performed under the configuration controls described above.

ii) Preventive Maintenance

Preventive maintenance (PM) will be performed to ensure that IROFS are available to perform their function and are reliable. PM tasks will be developed through a review of manufacturer recommendations, industry standards, and available historical operating information. Deviations from manufacturer recommendations or industry standards will be documented. The formal documented bases for the tasks will be developed, evaluated, and approved by the engineering organization. Changes to the tasks may be made as a result of feedback, corrective maintenance, and incident investigations to modify the frequency or scope of PM work.

iii) Surveillance and Monitoring

Surveillance and monitoring, consisting of performance checks, calibrations, tests and/or inspections, will be performed for IROFS at specified intervals to verify that their design function can be met. The surveillance program will adhere to 10 CFR 70.64, "Inspection, Testing, and Maintenance Baseline Design Criteria." The surveillances will be included in the work control process and the results will be trended. Frequencies will be adjusted or other corrective actions taken as necessary.

iv) Functional Testing

Post Maintenance Testing

To provide assurance that IROFS will perform their intended safety function following maintenance, a post-maintenance testing program will be established. The test requirements will be included in the work packages and the test results will be documented. The testing will confirm that the maintenance was satisfactorily performed, any identified deficiency corrected, and that the reliability of the IROFS was not adversely affected.

Control of Measuring and Test Equipment

A measuring and test equipment (MT&E) program will be developed to assure that MT&E are properly controlled, calibrated, and adjusted at specified time periods to assure the required accuracy, range, and stability. Available standards, traceable to the National Institute of Standards and Technology will be used. Where standards are not available, the engineering organization will approve and document the bases for calibration.

Equipment and Work History

The work control process will collect data from the performance of preventive and corrective maintenance activities to identify the need for modifications and improvements for the maintenance program and to improve the reliability of IROFS. The database will be maintained for historic purposes and used by the engineering organization to evaluate the reliability of IROFS.

C) Training and Qualifications

Program Policy

Section 11.3, "Training and Qualifications," of the application addresses the applicant's commitments for training and qualifications of the Lead Cascade personnel. It requires that those personnel who perform activities relied on for safety must have the applicable knowledge and skills necessary to design, operate, and maintain the facility in a safe manner. Personnel are tested as necessary to ensure that they are qualified on practices important to public and worker safety, safeguard of licensed material, and protection of the environment.

i) Training Program Organization and Administration

The applicant states that, during the refurbishment/start-up phase of the Lead Cascade project, the Training Manager is responsible for preparation, presentation, and documentation of employee orientations and for technical and qualification training program development and implementation. During facility operation, the Project Support Manager has the responsibility for training. Training personnel interface with the functional line managers to coordinate and assist in the design, development, implementation, and evaluation of the training and qualification programs in the functional areas. The Training Manager is responsible for establishing procedures governing the development and implementation of training programs. Performance Based Training (PBT) methodology is used for those tasks associated with the design, modification, operation, or maintenance of IROFS.

The Lead Cascade functional organization managers are responsible for defining the job-specific training needs and ensuring completion of training and qualification for personnel within their organization. Workers relied upon to design, operate, or maintain IROFS are trained and evaluated for qualifications prior to assignment of these duties. Task or duty area qualification is granted by line management based on successful evaluation of the worker's mastery of the learning objectives presented during training. Maintenance of qualification is contingent upon successful completion of continuing training and/or through satisfactory on-the-job training (OJT) evaluations.

Construction personnel, Lead Cascade operators, technicians, maintenance personnel, and other staff whose actions are relied upon for safety are required to complete the applicable training programs or have equivalent experience or training. Lead Cascade functional organization managers develop and maintain a description of each organization's training requirements. These requirements are identified in Training Requirement Matrices (TRMs) approved by the line and training management. Line management is responsible for placing work restrictions or removing employees from performing functions in areas where training is deficient.

Performance Based Training (PBT)

The applicant's training program for those personnel who design, operate, or maintain IROFS is based on a PBT methodology and includes the following elements:

1. Conduct of needs/job analysis and identification of tasks for training;

2. Development of learning objectives;

3. Development of lesson plans and training guides;

4. Evaluation of trainee mastery of learning objectives; and

5. Evaluation of the effectiveness of training.

ii) Conduct of Needs/Job Analysis and Identification of Tasks for Training

The applicant will use a needs/job analysis to identify the tasks affecting worker or public safety, safeguards of regulated material, or protection of the environment as identified in the ISA Summary. The analysis is conducted utilizing either written surveys or the tabletop method with subject matter experts. The training programs for personnel for these tasks are based on a needs/job analysis.

Procedure changes, equipment changes, job scope changes, facility modifications, and other changes affecting task performance are monitored and evaluated for their impact on the development or modification of initial and continuing training programs.

iii) Development of Learning Objectives, Lesson Plans, and Training Guides

The applicant has committed to the establishment of learning objectives to identify the training content and to define satisfactory trainee performance for the task or group of tasks selected for training. Learning objectives state the requisite knowledge, skills, and abilities the trainee must demonstrate. Learning objectives are documented in lesson plans and training guides and are revised as necessary based on changes in procedures, facility systems/equipment, or job scope.

Learning objectives are analyzed to determine the appropriate training setting. Classroom lesson plans, OJT guides, or other instructional materials are procured or developed based on this instructional analysis and design. Lesson plans and other training guides provide the guidance and structure necessary to ensure consistent delivery of training material from trainer to trainer and class to class.

iv) Evaluation of Trainee Mastery of Learning Objectives

The applicant states that within the job position/worker classification training programs are logical instructional blocks or "modules" which are presented in such a manner that specific learning objectives are accomplished. Trainee progress is evaluated by line and training management through a variety of performance demonstrations such as written examinations, oral examinations, and practical tests to ensure mastery of the job performance requirements or learning objectives contained in these modules. Remediation is provided as appropriate.

v) Evaluation of the Effectiveness of Training

Systematic evaluations of training effectiveness and its relation to on-the-job performance are

used to ensure that the training program conveys required skills and knowledge and to revise the training, where necessary, based on the performance of trained personnel in the job setting.

Line and training management conduct self-assessments and evaluations of the individual training programs. Nuclear Safety and Quality (NS&Q) auditors provide additional assessments through the audit program. These assessments and evaluations are used to determine training program strengths and weaknesses.

vi) Training Instructor/Developer Qualification

Training Instructor/Developer Qualification is the responsibility of the PORTS Training Organization. Training is provided to designated training personnel who develop and/or conduct classroom training and/or OJT evaluations for Lead Cascade personnel. The program consists of modules designed to train instructor/developers in the application of the PBT methodology. All training program materials are reviewed and approved by subject matter experts and line management prior to implementation.

vii) Maintenance of Training Records

Training attendance records, examinations, employee qualification records, and program needs are maintained in an accurate, auditable manner to document each employee's training. The individual employee training records are maintained by the PORTS Training Organization.

D) Procedures Development and Implementation

Procedures Process Policy

In license application Section 11.4, "Procedures," the applicant has stated that a management controls program has been established for the development, issuance, and control of Lead Cascade procedures. The applicant committed to the use of approved and controlled written procedures to conduct nuclear safety, safeguards, and security activities for the protection of the public, facility employees, and the environment. Procedures are used to ensure safe work practices and apply to workers, visitors, contractors, and vendors. Activities involving nuclear material and/or IROFS are conducted in accordance with approved procedures.

The procedure process utilizes a graded approach to provide the necessary rigor for safe Lead Cascade operation, assure the applicant's commitments to meeting regulations and standards, and assure a balance of effective safety with practical efficiency in facility operations.

Procedure Hierarchy and Types

The applicant's procedure hierarchy is established in four levels:

Level 1 - Policy statements issued by executive management that apply to Lead Cascade personnel.

Level 2 - Standard practice procedures that apply to more than one organization.

Level 3 - Procedures issued at the organization level that applies to more than one group within a larger group or specific organization.

Level 4 - Procedures issued within a group or sub-function.

The following types of procedures are used by the applicant in the Lead Cascade:

1. Administrative Procedures: Procedures that deal with policy or programs/administrative systems, provide programmatic requirements, and do not normally involve manipulation of equipment;

2. Operating (non-administrative) Procedures: Procedures that direct or cause operation/maintenance of equipment or may directly affect any physical characteristics of equipment;

3. Alarm Response Procedures: Procedures that provide information that identifies the symptoms of the alarm, possible causes, automatic actions, the immediate operator action to be taken, and the required supplementary actions;

4. Off-Normal Procedures: Procedures that describe actions to be taken during unusual or out-of-the ordinary situations; and

5. Emergency Operating Procedures: Procedures directing actions necessary to mitigate potential events or events in progress that involve needed protection of on-site personnel, the public health and safety, and the environment.

Procedure Process

The applicant states that procedures are developed or modified through a formal process incorporating the change controls described in Section 11.1, 'Configuration Management," of the license application. The procedure process utilizes nine basic elements to accomplish procedure development, review, approval, and control: identification, development, verification, review and comment resolution, approval, validation, issuance, change control, and periodic review. These elements are discussed in the following sections.

Identification

As a minimum, a procedure is required for:

1. The operation of IROFS and the management measures supporting those IROFS as identified in the ISA Summary;

2. Operator actions necessary to prevent or mitigate the consequences of accidents described in the ISA Summary; and

3. Safe work practices to control processes and operations with special nuclear material, IROFS, and/or hazardous chemicals incident to the processing of licensed material.

Any new or revised NRC requirements that are promulgated are evaluated to determine the impact on existing implementing procedures or to identify the need for new implementing procedures. Procedures are reviewed following unusual incidents to determine if changes are appropriate based on the root cause and corrective action determination for the particular incident.

Lead Cascade organization managers have the responsibility for identifying which tasks must be addressed in procedures within their areas of control.

Development

Procedure development and quality is the user organization's responsibility. Procedure development is accomplished in accordance with procedural guidance which requires that:

1. A system be in place to track and document the procedure process;

2. Interviews with procedure users and process walk-downs are utilized to ensure procedures are usable; reflect as-built conditions and process operations; and maintain management controls for nuclear safety, safeguards, and security;

3. The procedure use category is determined. This documents the designation of a procedure as In-Hand (Continuous Use), Reference Use, or Information Use. The designation is based on the administrative or non-administrative use of the procedure, and the safety or financial consequences of failing to adhere to procedural requirements; and

4. Operating and maintenance procedures will identify, as applicable, the purpose, the governing regulations, policies, and guidelines, hazards and safety considerations, operating limits, precautions regarding hazardous chemicals or special nuclear material, hold-points or safety checkpoints, IROFS and their functions, steps for each phase or activity, controls, testing, qualification, and records requirements.

Verification

Procedures are verified by the procedure owner/user during the procedure development/change process. Verification is made of the technical accuracy to ensure that all technical information including formulas, set points, and acceptance criteria are correctly identified in the procedure. An administrative verification is also made that verifies the procedure format and style and that it is consistent with the procedure writing guide.

Review

Draft, new procedures and procedure changes are distributed for technical reviews, safety discipline reviews (e.g., nuclear criticality, fire, radiation, industrial, and chemical process safety), and cross-discipline reviews, as needed.

Functional area and cross-discipline reviews are performed for the new procedure or procedure change. Comments/questions generated during the review process are resolved with the originating organizations. 10 CFR 70.72 and intent/non-intent screenings are performed for new and changed procedures (except minor administrative changes that are processed according to the procedure process).

Approval

Following the resolution of review comments, procedures are approved. Approval authority rests with the responsible manager. Managers ensure that appropriate training is completed on new and revised procedures in accordance with Section 11.3, "Training and Qualification," of the license application.

Validation

The purpose of procedure validation is to ensure that no technical errors or human factor issues were inadvertently introduced during the procedure review process. Validation is required for new procedures or for intent changes to the procedure. Validation is performed in the field by qualified personnel, and may be accomplished by detailed scrutiny of the procedure as part of a walk-through exercise or drill. If a particular system or process is not available for a walk-through validation, talk-through may be performed in the particular shop or training environment. Performance of procedure validation is documented.

Issuance and Distribution

Procedures are issued and controlled in accordance with the Records Management and Document Control (RMDC) program procedures. Copies of current approved procedures are made available to users via electronic and/or hard copy distribution in the work areas.

Temporary Changes

Temporary changes to procedures can be made, provided that the temporary change does not result in a change to the ISA as determined by the 10 CFR 70.72 review, the temporary change does not constitute an intent change (i.e., a change in scope, method, or acceptance criteria that has safety significance), and the change is documented.

Periodic Review

Approved procedures are periodically reviewed to ensure their continued accuracy and usefulness. Procedures are periodically reviewed according to established criteria. The review cycle is one year for emergency operating, alarm response and procedures dealing with highly hazardous chemicals, and five years for all other procedures. When conducting the periodic review, the procedure owner or subject matter expert performs a complete administrative and technical (requirements and references) review ensuring information is complete and accurate and that the procedure is usable as written.

Use and Control of Procedures

In-Hand (Continuous Use) procedures are followed step-by-step and are present in the work area while the task is being performed. In-Hand procedures, approved equipment alignment check sheets (e.g., valve lineups or electrical switching orders), or approved operator aids are developed for IROFS that: have extensive or complex tasks; are infrequently performed; or for which operations must be performed in a specified sequence.

Reference Use procedures are provided for routine procedural actions that are frequently repeated or of minimal complexity and can be performed from memory. Reference Use procedures are not required to be present in the work area.

Information Use procedures are followed to implement administrative or programmatic requirements.

If a step of any procedure cannot be performed as written, work is stopped, the system is immediately placed in a safe condition, and corrective actions are initiated, in accordance with facility procedures.

Records

Records generated during procedure use are identified in the governing procedure and controlled according to the facility RMDC program practices as described in Section 11.7 of the license application.

Activities Requiring Procedures

The applicant has identified the minimum functional and specific activities that are to be covered by written procedures. These include administrative procedures for training, internal audits and inspections, investigations and reporting, design records and document control, and facility changes; system procedures that address start-up, operation and shutdown; abnormal operation/alarm response procedures; procedures for maintenance activities; and emergency procedures.

E) Audits and Assessments

Audits and Assessments Policy

In Section 11.5, "Audits and Assessments," of the license application, the applicant commits to implementing a system of audits and assessments to help ensure that the health, safety, and environmental programs, as described in this license application, are adequate and effectively implemented. The system is designed to ensure comprehensive program oversight for all Lead Cascade activities and functions. Audits are conducted at least once every three years.

Audits

The applicant states that audits are conducted by qualified auditors in the Quality Services

Organization for the refurbishment/start-up phases and in the PORTS NS&Q Organization for the operations phase in accordance with written procedures or checklists. The auditing organizations are independent from operations of the Lead Cascade. Audits will verify the effectiveness of health, safety, and environmental programs and their implementation and determine the effectiveness of the process being assessed. Audits will further verify that Lead Cascade operations are being conducted safely in accordance with regulatory requirements and license application commitments.

These audits are conducted in accordance with Section 18 of the quality assurance program description (QAPD) and use written procedures or checklists. Audits are performed under the direction of a Lead Auditor, qualified in accordance with the American Society of Mechanical Engineering (ASME) NQA-1, Supplement 2S-3. Lead Auditors and staff auditors are functionally and organizationally independent of the programs and activities that are examined.

In addition to periodically evaluating aspects of the QAPD, the applicant states that the audits are to be conducted for the areas of radiation safety, nuclear criticality safety, chemical safety, fire safety, environmental protection, emergency management, quality assurance, CM, maintenance, training and qualification, procedures, incident investigation, and records management.

The applicant commits to documenting and reporting audit results to Lead Cascade senior management as specified in facility procedures. Provisions are to be made for reporting and corrective action, where warranted. A Lead Cascade Corrective Action Program, described in Section 11.6 of the applicant's license application, is administered by the Regulatory Organization to ensure proper control of corrective actions as defined in Section 16 of the QAPD.

Assessments

The applicant states that management, responsible for implementing portions of the QAPD, is required to perform assessments to verify the adequacy of the part of the QAPD for which they are responsible and to assure its effective implementation. Personnel from the area being assessed may perform the assessment, provided that they do not have direct responsibility for the specific activity being assessed. Results of assessments are documented and observations from these programmatic assessments are resolved by appropriate licensee management.

Organization managers maintain an assessment process within their organization to assess the adequacy, and effectiveness of the implementation of the programs under their cognizance. As a minimum, these assessments are conducted for the areas of radiation safety, nuclear criticality safety, chemical safety, fire safety, environmental protection, emergency management, quality assurance, CM, maintenance, training and qualification, procedures, incident investigation, and records management.

Assessment results are documented and reported as specified in Lead Cascade procedures. Provisions are made for reporting and corrective action, where warranted, in accordance with the Lead Cascade Corrective Action Program.

The staff's review has determined that the applicant has adequately:

1. Addressed policy directives covering the audit and assessment function, the activities to be audited, audit frequency, guidance in conducting the audit or assessment, assignment of responsibilities for each phase of the work, and procedures for recording the results and recommending actions to be taken;

2. Committed to conduct internal audits and independent assessments of activities significant to facility safety and environmental protection;

3. Committed to conduct audits to verify that operations are being conducted in accordance with regulatory requirements and commitments in the license application;

4. Committed to conduct independent assessments by off-site groups or individuals not involved in the licensed activity, to verify that the health, safety, and environmental compliance functions are effectively achieving their designed purposes;

5. Committed to conduct audits and assessments for the areas of radiation safety, nuclear criticality safety, chemical safety, fire safety, environmental protection, emergency management, QA, CM, maintenance, training and qualification, procedures, incident investigation, and records management; and

6. Committed to use qualified personnel to conduct audits and assessments, who do not have direct responsibility for the function and area being audited or assessed.

The staff positions and committees responsible for audits and assessments are specified. The levels of management to which results are reported, and the systems to provide corrective actions are also described.

F) Incident Investigations

Section 11.6 of the application describes the incident investigation process for the identification, reporting, and investigation of abnormal events. The process incorporates the reporting requirements of 10 CFR 70.50 and 70.74. The applicant has committed to establish procedures to assure that abnormal events and conditions are promptly reported to appropriate Lead Cascade personnel, assessed, and, when required, reported to NRC. The process will involve reporting to line management or the Plant Shift Superintendent (PSS). The PSS will assess and categorize the event by procedure using the notification and reporting criteria of 10 CFR 70.50 and 70.74.

The level of investigation will be based on a graded approach relative to the severity of the event and conducted and documented in accordance with procedures. Each reportable event where a follow-up written report to the NRC is required will be investigated to determine the root cause and corrective actions necessary to prevent recurrence. Other events not requiring a written report will be evaluated using the Corrective Action Program to determine actions to be taken.

A prompt, risk-based evaluation will be performed, and an investigation will be initiated within 48-hours of the event, or sooner, depending on the safety significance. A procedure will provide

a documented plan for conducting the investigation. A reasonable, systematic, structured approach will be used to determine the specific or generic root causes of the event. A record of IROFS failures will be maintained as part of the investigation and updated in accordance with regulations. Auditable records and documentation related to the investigation will be kept for at least two years or the life of the operation, which ever is longer, at any location. The original investigation reports will be available to the NRC upon request.

The investigators will be independent of the line function involved in the incident under investigation and have the authority to obtain all of the information necessary to conduct the investigation. Line management will fully cooperate with the investigators. The individual leading the investigation will be trained and qualified in root cause analysis techniques. If a team is used, at least one member will be a process expert.

The applicant has requested an exemption from 10 CFR 70.50(c)(2) by extending the 30-day event reporting requirement to 60 days. This exemption has been granted to United States Enrichment Corporation (the applicant) for the Portsmouth Gaseous Diffusion Plant (PORTS) certificate. This exemption provides the aplicant more time to complete its event investigation, which results in more meaningful investigation. The NRC staff finds this request reasonable and appropriate for the Lead Cascade. For significant conditions adverse to quality or when a follow-up written report to the NRC is required, corrective actions will be developed, tracked in a database, and monitored through completion. Corrective actions will be taken within a reasonable time, commensurate with safety significance. Record revisions will be made promptly and evidence files to support action closure will be maintained. Details of the event will be compared with the accident sequences already considered by the ISA. When necessary, the ISA Summary will be updated and relevant findings will be reviewed by Lead Cascade personnel.

G) Records Management

i) Records Management Policy

Section 11.7, "Records Management and Document Control," of the license application states that:

Records Management and Document Control (RMDC) programs are established to ensure records and documents required by the QAPD are appropriately managed and controlled. These programs are designed to meet the specific record keeping requirements set forth in 10 CFR Part 70 and the applicable provisions of other parts of 10 CFR. These programs provide administrative controls that establish standard methods and requirements for collecting, maintaining, and disposing of records. The administrative controls for the generation and revision of records and documents are contained in implementing procedures.

ii) Records Management Program

The applicant has committed to a records management program that provides direction for the handling, transmittal, storage, and retrievability of records. The records management program design is to provide for adequate assurance that the appropriate records of IROFS are maintained

in accordance with the baseline design criteria (BDC) contained in CFR 70.64(a) and the defense-in-depth requirements of 10 CFR 70.64 (b). Records, such as letters, drawings, and specifications, must include all pertinent information, such as stamps, initials, and signatures. Records are categorized and handled in accordance with their relative importance to safety and storage needs. Special provisions are made for handling contamination and ensuring their inclusion in the program. This program is implemented through procedures that provide guidance for the following program requirements:

1. Documents designated to become records must be legible, accurate, complete, and contain an appropriate level of detail commensurate with the work being performed and the information required for that type of record;

2. Records will clearly and specifically identify the items or activities to which they apply;

3. Records will be authenticated or validated by the manager of the organization that originates the record, or his designee, as specified in the procedure which controls the generation and revision of these records;

4. Methods are specified for indexing, filing, and locating records within the record system to ensure the records can be retrieved in a timely manner;

5. Records retention times are specified in a retention schedule, developed by the manager of the organization that originates the record, or the designee. The process for disposition of records that have reached the end of their retention lifetime is specified by procedures and conforms to applicable requirements;

6. Corrections to records are approved by the organization that created the record unless other organizations are specifically designated. Changes are made by clearly indicating the correction, the date of the correction and the identification of the individual making the correction;

7. Controls are established for protection of records from deterioration, loss, damage, theft, tampering, and/or unauthorized access for the life of the record. Requirements include instructions on protection of records by the record originator until they are transferred to Records Management. Instructions for the protection of special record media such as radiographs, photographs, negatives, microform and magnetic media are provided to prevent damage from excessive light, stacking, electromagnetic fields, temperature, humidity, or any other condition adverse to the preservation of those records. Records, which cannot be duplicated, are stored in a fashion that minimizes deterioration; and

8. Records encompassed by the QAPD are stored in authorized facilities or containers providing protection from fire hazards, natural disasters, environmental conditions, and infestations of insects, mold, or rodents. Storage facilities are maintained to ensure continuous protection of the records. Requirements are specified for both permanent and temporary storage of records.

For "Permanent Storage," records are permanently stored in facilities satisfying the following requirements:

1. Storage in 2-hour-rated containers meeting National Fire Protection Association (NFPA) 232-1986 or NFPA 232 AM-1986 or both, with the clarification that if the NFPA 232 (or 232 AM) method of storage in 2-hour-rated containers is used, any exceptions to this standard will be documented and justified by the authority having jurisdiction; or

2. Storage of duplicate copies in separate facilities that are sufficiently remote from each other to eliminate the possibility of exposure to simultaneous hazards; or

3. Storage in facilities that have the following: doors, structures, frames, and hardware that comply with a minimum 2-hour fire rating; a fire protection system; 2-hour fire rated dampers on all boundary penetrations; sealed floor surface to minimize concrete dust; adequate access and aisle ways; and a prohibition on eating, drinking, or smoking and performing work other than that associated with records storage or retrieval.

For temporary storage, the RMDC process requires that those completed records documenting nuclear safety or safeguards and security matters, that are being held temporarily by originating organizations, be properly protected by maintaining them in 1-hour, fire-rated containers. If 1-hour fire-rated containers are used, they either bear an Underwriters Laboratory label (or equivalent) certifying 1-hour fire protection, or the containers are certified for 1-hour fire protection by an authorized individual competent in the field of fire protection. Procedural requirements are used to limit the length of time during which records may be maintained in temporary storage, based on the significance of the record.

A record transmittal process is used to formally transmit records to Records Management. The process includes a receipt acknowledgment that notifies the sending organization that the records have been received and accepted.

Requirements for controlling access to records and maintaining accountability for records are provided to ensure that only authorized personnel have access to records and to prevent loss, damage, or inadvertent destruction of records.

Records management requirements for goods or services procured from outside suppliers are specified in the applicable procurement documents.

Examples of the records required by 10 CFR Parts 19, 20, 21, 25, and 70 are identified in Section 11.7.5 of the license application. Computer programs used in the records management program are controlled and maintained in accordance with the PORTS "Computing and Telecommunications Security Manual" and Information Systems procedures. These requirements and practices provide for virus protection as well as access control to the Records Management program database and ensure continuing usability of the codes as hardware and software technology change. Routine backups of the records management database are made. Precautions are taken to ensure that computer data that constitute a record are stored in a format that is readily retrievable even as hardware and software technology evolve. The storage format

of computer data is reviewed as required to determine threats to future retrievability, and if necessary, the data are translated to an updated format and verified acceptable.

The overall effectiveness of the Records Management program is evaluated through the audit program described in the Section 18 of the QAPD. Deficiencies identified are corrected in a timely manner in accordance with Section 11.6 of the license application.
The Project Support Manager is responsible for the Lead Cascade RMDC program. Appropriately trained and qualified personnel manage the RMDC programs. General training in RMDC is provided to employees as part of the general employee training, as described in Section 11.3 of the license application.

H) Other QA Elements

i) QA Policy

The applicant has developed QA principles that apply to the design, fabrication, refurbishment, modification, testing, operation, and maintenance of the Lead Cascade facility. These principles are described in the USEC Gas Centrifuge Quality Assurance Program Description (QAPD), Revision 0, the final version which was submitted to the NRC on June 19, 2003, and is referenced in Section 11.8, "Other QA Elements," of the Lead Cascade license application. The applicant does not commit to any specific consensus or QA programmatic standard; however, the QAPD does address each of the 18 standard criteria and requirements of ASME NQA-1, "QA Requirements for Nuclear Facility Applications" (NQA-1). The staff reviewed the applicants' QAPD for adequate application of QA elements to IROFS in accordance with the guidance and criteria in NUREG-1520. The applicant's QAPD elements are structured to apply appropriate measures to IROFS in proportion to the importance of the item to safety (a graded approach).

ii) Organization

Section 1 of the QAPD describes the (a) organizational structure, (b) functional responsibilities, and (c) charts of the lines, interrelationships, and areas of responsibility and authority for all organizations performing activities relied on for safety. These are provided as organization charts and position responsibility descriptions for the Lead Cascade design, refurbishment, and start-up phases and for the operations phase. As the operator of the Lead Cascade, the applicant maintains overall responsibility for design, fabrication, refurbishment, testing, operation, and modification. The applicant's Senior Vice President has overall responsibility for the Lead Cascade, and is also responsible for the QA program and for determining the status, adequacy, and effectiveness of the QAPD. The Nuclear Safety and Quality Manager reports to the Senior Vice President and has independent oversight responsibility for implementation of the QAPD. Positions and organizations responsible for ensuring that appropriate QA has been established and for verifying that activities affecting quality have been correctly performed have sufficient authority, access to work areas, and organizational independence to carry out their responsibilities. Organizational responsibilities for engineering, design, procurement, maintenance, operations, audits and assessments are identified.

iii) QA Program

The applicant described its application of QA elements for IROFS in Section 2, "QA Program," of the QAPD. The applicant has committed to a graded approach to QA, in which measures are implemented consistent with an item's importance to safety. The application of QA elements is documented, planned, implemented, and maintained to provide reasonable assurance that, together with the other management measures, IROFS will be available and reliable when needed. The QAPD is applied to the facility, structures, processes, systems, equipment, components, computer programs, and activities of personnel, in accordance with their QL classification. The QLs are established as follows:

QL-1 A single control or IROFS that prevents or mitigates a high consequence event.

QL-2 A control that is one of two or more controls or IROFS that prevents or mitigates a high consequence event; or any control or IROFS that prevents or mitigates an intermediate consequence event.

QL-3 Any item other than QL-1 and QL-2; QL-3 items are controlled in accordance with standard commercial practices.

Implementing procedures provide for graded application of QA elements taking into consideration:

1. QL (risk significance);

2. Applicable regulations, industry codes and standards;

3. Complexity or uniqueness of an item or activity and the environment in which it is to function;

4. Quality history of the item in service;

5. Degree to which functional compliance can be demonstrated or assessed by test; inspection or maintenance methods;

6. Anticipated life span;

7. Degree of standardization;

8. Importance of data generated;

9. Reproducibility of results; and

10. Consequences of failure.

The results of the application of the graded approach to quality are incorporated into design requirement documents, specifications, procedures, instructions, drawings, inspection plans, procurement documents, and other documents that establish the requirements for items and activities.

The QA program commitments include the personnel training, testing, experience or qualification requirements of American Society for Nondestructive Testing Recommended Practice No. SNT-TC-1A, June 1980 Edition, Supplement 2S-4, "Supplementary Requirements for Personnel and Training," of Part I of NQA-1- 1994, Supplement 2S-1, "Supplementary Requirements for the Qualification of Inspection and Test Personnel," Part 1 of NQA-1-1994, and Supplement 2S-3, "Supplemental Requirements for the Qualification of QA Program Audit Personnel," of NQA-1-1994.

iv) Design Control

Section 3 of the QAPD describes the applicant's design control process and procedures that include control design inputs, process, analyses, verification, interfaces, changes, and design documentation and records. Approved procedures provide for performing the design in a planned, controlled, and documented manner. The design control process is also applied to the ISA and management measures. Design inputs, such as design bases, performance requirements, regulatory requirements, codes, and standards, are identified and documented as design requirements. Design methods, materials, parts, equipment, and processes that are essential to the function of the IROFS are selected and reviewed for suitability of application. Final design output documents, including changes, can be related to the design input to permit verification. Design outputs that are computer programs are developed, validated, and managed in accordance with Basic Requirement 11, "Test Control," of NQA-1-1994 and Part II, Subpart 2.7, "QA Requirements for Computer Software for Nuclear Facility Applications." The applicant states that internal and external design interfaces are identified and controlled and design efforts are coordinated among participating organizations. Design information transmitted across interfaces is reviewed, approved, documented, and controlled. Final design documentation and records that provide evidence that the design and design verification processes were performed in accordance with the QAPD are collected, stored, and maintained.

v) Instructions, Procedures, and Drawings

In Section 5 of the QAPD, the applicant states that activities affecting quality are prescribed by and performed in accordance with documented instructions, procedures, and drawings of a type appropriate for the circumstances. These documents include or reference appropriate acceptance criteria for determining that prescribed activities have been satisfactorily accomplished. The QAPD establishes the policy requirements for these documents. The applicant's QA organization reviews QA implementing procedures for compliance and consistency with this QAPD and ensures that the QAPD provisions are effectively incorporated into QA implementing procedures.

vi) Document Control

The applicant's commitments and provisions for the application of the document control QA element are addressed in QAPD Section 6, "Document Control." The preparation, issuance, and modification of documents that specify quality requirements or prescribe activities affecting quality are controlled to provide assurance that the appropriate documents are in use. Document changes are reviewed for adequacy and approved for implementation by authorized personnel. Procedures and instructions assure that documents are prepared; reviewed for adequacy, correctness and completeness by a qualified individual; approved for release by authorized personnel; distributed to the location where the activity is performed prior to commencing work; and used in performing the activity. The applicant's procedures require the creation and maintenance of a controlled document index to track and control approved revision levels of these documents. Changes to documents are reviewed and approved in the same manner as the original unless other organizations are specifically designated.

vii) Procurement Control

The applicant's commitments and provisions for control of the procurement process, procurement documents, and procured material, components, and services are described in QAPD Section 4, "Procurement Document Control," and Section 7, "Control of Purchase Items and Services." The applicant states that applicable design bases and other requirements necessary to provide reasonable assurance of quality are included or referenced in documents for procurement of items or services relied on for safety. To the extent necessary, suppliers are required to have acceptable QA programs or systems consistent with the applicable QAPD sections and the quality level of the item or service to be procured. Purchased IROFS and services relied on for safety are controlled to provide reasonable assurance of conformance with specified requirements.

viii) Identification and Control of Items

In QAPD Section 8, "Identification and Control of Items," the applicant states that controls are established to identify and control items from initial receipt and fabrication of the items up to and including installation and use to provide assurance that incorrect or defective items are not used or installed. Physical identification is used to the maximum extent possible. Traceability of items to specific records is provided when specified by codes, standards, or specifications. Items with a limited shelf or operating life are identified and controlled and procedures provide for item identification consistent with the planned duration and conditions of storage.

ix) Control of Processes

In QAPD Section 9, "Control of Processes," the applicant describes its commitments for control of processes affecting quality of items and services. Requirements are established to maintain the acceptability of special processes, such as those used in welding, heat treating, and nondestructive examination, and to assure that they are performed by qualified personnel using qualified procedures and equipment. Certification of individuals, qualification of processes, and/or control of process parameters, equipment, calibration or acceptance criteria will be

prescribed when necessary. Records will be maintained of currently qualified personnel, processes, and equipment for special processes.

x) Inspection

The applicant's commitments for the QA element of inspection are described in QAPD Section 10, "Inspection." The applicant commits to planned inspections to verify conformance of items to applicable requirements. Inspection requirements are based on the importance of the item or activity, and are specified in written procedures with provisions included for documenting and evaluating inspection results. Personnel qualification programs are established for inspection and test personnel. Inspection records are to contain information related to the item, inspector, type of observation or inspection plan, results of the inspection or acceptability of the item, and actions taken with regard to any non-conformances.

xi) Test Control

The applicant stated in QAPD Section 11, "Test Control," that tests are performed to verify conformance to specified requirements, to demonstrate satisfactory performance, or to collect data. These tests are to include design verification tests, acceptance tests, pre-operational tests, post-maintenance tests, and operational tests. Test requirements are specified in written procedures with provisions included for documenting and evaluating test results. Test records are to contain the item tested, test date, tester, type of observation, test procedure, results and acceptability, actions taken on any deviations, and person evaluating the results.

xii) Control of Measuring and Test Equipment

Section 12 of the QAPD, "Control of Measuring and Test Equipment," describes the applicant's controls to provide reasonable assurance that tools, gauges, instruments, and other measuring and testing devices are properly identified, controlled, calibrated, and adjusted at specified intervals to maintain performance within required limits. This section appropriately specifies requirements for calibration procedures, a control system for items requiring calibration, application and actual item use documentation, out-of-calibration actions, and records.

xiii) Handling, Storage, and Shipping

QAPD Section 13, "Handling, Storage, and Shipping," describes the applicant's measures to control the handling, storage, shipping, cleaning, and preservation of material and equipment, in accordance with design and procurement requirements and work and inspection instructions to prevent loss, damage, or deterioration. Special covering, equipment, and protective environments are required, verified, and monitored where necessary for the protection of particular items. Special handling tools and equipment are provided, controlled, and maintained, and operators are trained or appropriately experienced. Special handling, preservation, storage, cleaning, packaging, or shipping instructions are established and used when essential to maintain acceptable quality.

xiv) Inspection, Test, and Operating Status

The applicant's provisions for inspection, test, and operating status are described in QAPD Section 14, "Inspection, Test, and Operating Status." These commitments are for control of test activities for IROFS to prevent inadvertent use of nonconforming items or bypassing of inspections and tests. The applicant states that procedures are established to control these activities to ensure that required inspections and tests are performed, and to ensure that items that have not passed the required inspections or tests are not installed, used, or operated. Status indicators are utilized when required, and authority for the application and removal of status indicators is specified.

xv) Control of Nonconforming Items

QAPD Section 15, "Control of Nonconforming Items," specifies the applicant's provisions to control the identification, segregation, disposition, and prevention of installation or use of nonconforming items. Nonconforming items are identified and segregated or measures are employed to prevent use of the item. Nonconforming items are reviewed and appropriately dispositioned and documented. The responsibility and authority for the evaluation and disposition of nonconforming items is defined and technical justification for the disposition is documented and subject to design control measures when applicable. Repaired or reworked items are re-examined in accordance with the original acceptance criteria or alternate disposition/acceptance criteria. Documentation of the nonconformance includes its description, disposition, and disposition signatures.

xvi) Corrective Actions

QAPD Section 16, "Corrective Action," states that conditions adverse to quality are identified and corrected as soon as practical. For significant conditions adverse to quality, the cause of the condition is determined and corrective action is taken to preclude recurrence. These actions are documented, reported to appropriate levels of management, and follow-up action taken to verify implementation of the corrective action.

xvii) QA Records

QAPD Section 17, "QA Records," describes the applicant's QA records system for the identification, retention, retrieval, and maintenance of records that furnish evidence of the control of quality for IROFS. Requirements for lifetime and nonpermanent records are identified. Custodianship responsibility is specified. A commitment is stated that the storage facilities for hard copy and microfilm lifetime records meet the requirements of NQA-1-1994, Supplement 17S-1, "Supplementary Requirements for QA Records," Section 4.4. For electronic records storage, backups or duplicate files are generated.

xviii) Audits

QAPD Section 18, "Audits," identifies the applicant's commitments for planning and scheduling internal and external audits to verify compliance with, and to determine the effectiveness of, QA.

Responsibilities and procedures are identified for assessing, auditing, documenting, and reviewing results and for designating management levels to review audit results; and provisions are made for review of audit findings, follow-up action, and documentation. The applicant will perform audits to ensure comprehensive program oversight at least once every three years.

xix) Provisions for Changes

QAPD Section 19, "Provisions for Changes," describes the applicant's provisions for continuing QA reviews and updates of QA documents based on reorganizations, revised activities, lessons learned, changes to applicable regulations, and other QA program changes. QAPD changes are governed by approved procedures and are controlled in accordance with 10 CFR 70.72. "Facility Changes and Change Process."

11.4 EVALUATION FINDINGS

A) Configuration Management (CM)

The applicant provided an appropriate description of the overall CM program that covered CM policy, design requirements, document control, change control, assessments, and design verification. The proposed CM program meets the requirements of 10 CFR 70.72 and provides adequate assurance that facility changes are identified and controlled. IROFS are required to be identified and documented, organizational responsibilities are identified and defined, and administrative controls, procedures and policies will be established to maintain the design configuration, and existing SSCs will be incorporated into the Lead Cascade Baseline Configuration.

B) Maintenance

The applicant has described the maintenance program policy and procedures, organizational responsibilities, qualification and training requirements, and work control process. The program elements related to surveillance/monitoring, corrective maintenance, preventive maintenance, and functional testing are adequately described. The staff concludes that the applicant's maintenance activities will provide reasonable assurance that the credited IROFS will be maintained in an available and reliable state when needed.

C) Training and Qualifications

Based on its review of the license application and the ISA Summary, and comparison of the licensee commitments for training and qualification to the review acceptance criteria guidance in NUREG-1520, the NRC staff has concluded that the applicant has adequately described and assessed its personnel training and qualification in a manner that (1) satisfies regulatory requirements and (2) is consistent with the guidance in NUREG-1520.

There is reasonable assurance that implementation of the described training and qualification commitments will result in personnel who are qualified and competent to design, refurbish, start up, operate, maintain, and modify the facility safely. The staff concludes that the applicant's plan for personnel training and qualification meets the requirements of 10 CFR Part 70.

D) Procedures

Based on the review of the application and comparison to the review acceptance criteria in NUREG-1520, the applicant has described a suitably detailed process for the development, approval, and implementation of procedures. The staff concludes that the applicant's commitments and plan description for procedures meets the requirements of 10 CFR Part 70.

E) Audits and Assessments

Based on the review of the license application, the NRC staff has concluded that the applicant has adequately described its audits and assessments. The staff has reviewed the applicant's commitments and description of its policy directives, plans and procedures for audits and assessments and finds it acceptable.

The staff concludes that the applicant's plan for audits and assessments meets the requirements of 10 CFR Part 70 and provides reasonable assurance of protection of the health and safety of the public, workers and the environment.

F) Incident Investigations

The applicant has committed to establish a process for the identification, reporting and investigation of abnormal events, commensurate with their importance to safety. Responsibilities are outlined and the program includes provisions for prompt safety evaluations, the conduct and documentation of formal investigations, the maintenance of records, and comparison of event sequences with the ISA Summary. The staff concludes that the applicant has adequately described and committed to conduct an incident investigation program to promptly identify and correct specific and generic root causes of abnormal conditions and events in a graded manner. The applicant's approach for incident investigations is thus considered acceptable.

G) Records Management

The staff has reviewed the applicant's records management system against the acceptance criteria in NUREG-1520 and concluded that the system: (1) will be effective in collecting, verifying, protecting, and storing information about the facility and its design, operations, and maintenance and will be able to retrieve the information in readable form for the designated lifetimes of the records; (2) will provide a records storage area(s) with the capability to protect and preserve health and safety records that are stored there during the mandated periods, including protection of the stored records against loss, theft, tampering, or damage during and after emergencies; and (3) will provide reasonable assurance that any deficiencies in the records management system or its implementation will be detected and corrected in a timely manner.

The staff concludes that the applicant's records management commitments and descriptions meet the requirements of 10 CFR 70.62(d), and provide reasonable assurance of adequate protection.

H) Other QA Elements

Based on its review of the license application and QAPD, the NRC staff has concluded that the applicant has adequately described the application of other QA elements to IROFS. The staff also concludes that:

1. The applicant has established and documented a commitment to an organization responsible for developing, implementing, and assessing the management measures for providing reasonable assurance of safe facility operations in accordance with the criteria in NUREG-1520.

2. The applicant has established and documented a commitment in the license application to QA elements, and the administrative measures for staffing, performance, assessing findings, and implementing corrective actions.

3. The applicant has developed a process for preparation and control of written administrative plant procedures, including procedures for evaluating changes to procedures, IROFS, and tests. A process for review, approval, and documentation of procedures will be implemented and maintained.

4. The applicant has committed to establish and document surveillances, tests, and inspections to provide reasonable assurance of satisfactory performance of IROFS to meet the requirements of 10 CFR Part 70, Subpart H. Specified standards or criteria and testing steps will be reviewed by the NRC as part of its readiness and management measures review, which will occur after issuance of any license but before introduction of UF_6 in the Lead Cascade facility.

5. Periodic independent audits are conducted to determine the effectiveness of the management measures. Management measures will provide for documentation of audit findings and implementation of corrective actions.

6. Training requirements have been established and documented to provide employees with the skills to perform their jobs safely. Management measures have been provided for evaluation of the effectiveness of training against predetermined objectives and criteria.

7. The organizations and persons performing QA element functions have the required independence and authority to effectively carry out their QA element functions without undue influence from those directly responsible for process operations.

8. QA elements adequately cover the IROFS, as identified in the ISA Summary.

Accordingly, the staff concludes that the applicant's application of other QA elements meets the requirements of 10 CFR Part 70 and provides reasonable assurance of protection of public and worker health and safety and of the environment.

The applicant has not established its management measures for IROFS. It is expected that these

would be established between the time of issuance of any license and introduction of UF_6 in the Lead Cascade facility. As such, any license issued for the Lead Cascade facility cannot become effective until the NRC has completed its review of the management measures. In addition, the NRC would also be conducting operational readiness inspections and reviews and other pre-operational inspections and observations to ensure that the facility is constructed and can be operated safely and securely in accordance with the requirements of the license. Only after the NRC determines the applicant's management measures, facility construction, and preoperational activities to be acceptable, would the license become effective. The NRC would then notify the applicant of this in writing. To address these issues, the applicant proposed the following license condition:

> "Nuclear operations may not commence until the Commission completes an operational readiness and management measures verification review to verify management measures have been implemented and confirms that the facility has been constructed in accordance with the requirements of the license. The applicant shall provide the Commission with 60 days advance notice of its plan to commence nuclear operations."

The NRC finds this condition to be acceptable.

11.5 REFERENCES

11.5.1 American Society of Mechanical Engineering. *Quality Assurance Requirements for Nuclear Facility Applications.* ASME NQA-1-1994. New York, NY: American Society of Mechanical Engineers. 1994.

11.5.2 Nuclear Regulatory Commission (U.S.) (NRC). NUREG-1520, "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility." NRC: Washington D.C. March 2002.

12.0 MATERIALS CONTROL AND ACCOUNTABILITY

The Nuclear Regulatory Commission (NRC) staff's review of the applicant's material control and accountability (MC&A) program is documented in Appendix D. The review contains information that has been marked as "Proprietary Information" by the applicant, pursuant to 10 CFR 2.790.

The staff concluded that the applicant provided an acceptable Fundamental Nuclear Material Control (FNMC) Plan for the Lead Cascade facility that will meet or exceed the applicable 10 CFR Parts 70 and 74 requirements to control and account for special nuclear materials that the United States Enrichment Corporation (the applicant) will use, possess, or has access to. As a result, the staff determined that the applicant meets the requirements in the area of MC&A to operate the Lead Cascade facility under 10 CFR Part 70.

13.0 PHYSICAL PROTECTION

The Nuclear Regulatory Commission (NRC) staff's review of the applicant's physical protection plan is documented in Appendix E. The review contains information that has been marked as "Proprietary Information" by the applicant, pursuant to 10 CFR 2.790.

The applicant's Physical Protection Plan for the Lead Cascade facility has been reviewed by the NRC staff. The methods and procedures outlined in the plan satisfy the performance objectives, systems capabilities, and reporting requirements specified in 10 CFR 73.67(a), 73.67(f), and 73.71. 10 CFR 73.74 is not applicable because the applicant will not import shipments of special nuclear material of low strategic significance (SNM-LSS.) The Physical Protection Plan for the Lead Cascade facility is acceptable and meets the NRC requirements for physical protection of SNM-LSS.

The NRC staff reviewed the applicant's Classified Matter Plan. On the basis of the review, the NRC staff concludes that the Classified Matter Plan satisfies the requirements of 10 CFR Part 95 and is acceptable for implementation.

14.0 SAFETY EVALUATION REPORT PREPARERS

The individuals listed below are the principal contributors to the preparation of this Safety Evaluation Report.

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